UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OCT 2 6 1983

MEMORANDUM FOR: Darrell G. Eisenhut, Director
Division of Licensing

THRU: Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

FROM: Mary F. Haughey, Project Manager
Licensing Branch No. 2
Division of Licensing

SUBJECT: NRR INPUT TO SALP - NINE MILE POINT 2

The NRR SALP input for Nine Mile Point Nuclear Station, Unit 2 covering the period 10/01/82 - 09/30/83 is provided as the attachment to this memo. The findings in the report are based on comments from the project manager and reviewers who have had interactions with the licensee. No additional comments were received from NRR Division Directors in review of the draft input.

Mary F. Haughey
Mary F. Haughey, Project Manager
Licensing Branch No. 2
Division of Licensing

Attachment:
As stated

cc: Region I

8311140510 XA

K/1

Significant licensing activities during this period include:

- Operating license application and acceptance review
- Completion of the revetment ditch review
- Caseload Forecast Panel, February 22-24, 1983
- Responses to acceptance review requests for information
- Management meeting on April 26, 1983
- Meeting on deviations from the Standard Review Plan, September 1, 1983
- A number of meetings and conference calls to discuss technical issues related to the safety and environmental reviews.
- Environmental Site Visit August 1-2, 1983

a. Management Involvement and Control in Assuring Quality

b.

c. Responsiveness to NRR Initiatives

d. Staffing

e. Safeguards

Conclusion

OCT 2 1985

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. B. G. Hooten
Executive Director
Nuclear Operations
c/o Miss Catherine R. Seibert
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

SUBJECT: EXAMINATION REPORT NO. 50-410/85-15 (OL)

This transmits the Examination Report of Operator Licensing Examinations conducted by USNRC Region I at the Nine Mile Point Unit 2 Facility the week of June 10, 1985. At the exit interview held on June 20, 1985, the preliminary results of these examinations were discussed.

The high failure rate of the reactor operator candidato causes us concern regarding the effectiveness of your licensed operator training program. You should review your program, the weaknesses noted in the enclosed report and address the causes of these failures. The NRC will be conducting programmatic and performance based inspections of your licensed operator training programs in conjunction with the near term operating license review process. The results of these reviews, along with the future results of NRC administered operator licensing examinations, will be considered during evaluation of your facility's readiness for fuel load and subsequent operation.

No reply to this letter is required. Your cooperation in this matter is appreciated.

Sincerely,

Original Signed By:

Richard W. Starostecki, Director
Division of Reactor Projects

Enclosure:
Examination Report No. 50-410/85-15 (OL) w/attachments 1, 2, 3

A-3

OFFICIAL RECORD COPY

OL NMP RPT - 0001.0.0
10/01/85

8510170295 PDR 2pp

OCT 2 1985

Docket No. 50-41J

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Enclosure:
Examination Report No. 50-410/85-15 (OL) w/attachments 1, 2, 3

A-3

8510170295 PDR 2pp

OFFICIAL RECORD COPY

OL NMP RPT - 0001.0.0
10/01/85

cc w/enclosure and attachments 1, 2, 3:
R. Abbott, Station Superintendent
R. Zollitsch, Plant Training Manager
Senior Resident Inspector
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
State of New York

cc w/o attachments 1, 2, 3 to enclosure:
Troy B. Conner, Jr., Esq.
D. Quamme, NMP-2 Project Director
C. Beckham, NMPC QA Manager
Department of Public Service, State of New York
W. Drew, Technical Superintendent
NRC Resident Inspector

bcc w/o attachments 1, 2, 3 to enclosure:
DRP Section Chief
Chief Examiner
Chief, AB/DHFS, NRR
OL File 12.0
Region I Docket Room (w/concurrences)
Master Exam File
J. Grant, DRP
Region, SLC

DRP:RI FCrescenzo 10/1/85 9/ /85	DRP:RI JBerry ~~9/ /85~~ 10/1/85	DRP:RI RKeller 9/ /85 10/1/85	DRP:RI [illegible]nville ~~9/~~ / /85 10	DRP:RI SCollins 10 ~~9/~~ \ /85
DRP:RI HKister 10/2 /85	DRP:RI Starostecki 10/2 /85			

OFFICIAL RECORD COPY OL NMP RPT - 0002.0.0
10/01/85

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION REGION I
OPERATOR LICENSING EXAMINATION REPORT

EXAMINATION REPORT NO. 85-15 (OL)

FACILITY DOCKET NO. 50-410

FACILITY LICENSE NO. CPPR-112

LICENSEE: Niagara Mohawk Power Corporation
300 Eric Boulevard West
Syracuse, New York 13202

FACILITY: NINE MILE POINT 2

EXAMINATION DATES: June 11-19, 1985

CHIEF EXAMINER: Paul J. Lemnge 10/2/85
Reactor Engineer (Examiner) date

REVIEWED By: Dave Lange 10/2/85
Lead Reactor Engineer (Examiner) date

REVIEWED By: fa D R Johnson 10/2/85
R. Keller, Chief date
Reactor Projects Section 1C

APPROVED By: HB Kister 10/2/85
H. Kister, Chief date
Reactor Projects Branch No. 1

SUMMARY: Operator licensing examinations were conducted at Nine Mile Point Unit 2 during the period June 11-19, 1985. Twelve Reactor Operator candidates and 12 Senior Reactor Operator candidates were administered written, oral, and simulator examinations. Nine Reactor Operator candidates failed the written examination; one Senior Reactor Operator candidate failed both the oral and simulator examination; one Senior Reactor Operator candidate failed the simulator examination.

REPORT DETAILS

TYPE OF EXAM: Initial

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written Exam	3/9	12/0
Oral Exam	12/0	11/1
Simulator Exam	12/0	10/2
Overall	3/9	10/2

1. Chief Examiner at Site: Frank Crescenzo

2. Other Examiners: John Berry, Dave Lange, Robert Keller William Cliff (PNL), Gary Sly (PNL).

3. Summary of generic strengths or deficiencies noted on oral exams:

   No outstanding generic weaknesses were noted; however, the following weaknesses were noted among some candidates:

   a. Knowledge of refueling operations and supervision of refuel floor activities was weak.

   b. Candidates were weak on noting abnormal conditions which have no annunciators, i.e., containment isolations, half scram on loss of RPS MG set, EHC oscillations.

   Strengths were noted in the following areas:

   a. SROs performed well on emergency classifications and notification.

   b. Most candidates performed well during plant walkthroughs.

   c. Knowledge and use of Technical Specifications.

d. Concern for personnel safety.

4. Summary of generic strengths or deficiencies noted from grading of written exams:

The Senior Reactor Operator candidates performed well on the written examination as evidenced by a 100% pass rate. The Reactor Operator candidates performed poorly on the written, as evidenced by a 25% pass rate with a high total score of 81.7%. It is difficult to identify specific generic weaknesses from grading of the RO examinations since all scores, both sectional and total, were, in general, low. In addition, all candidates who failed had failing total scores and at least one failing sectional score.

5. Personnel present at exit interview:

NRC Personnel

D. Lange, Reactor Engineer (Examiner)
F. Crescenzo, Reactor Engineer (Examiner)

Facility Personnel

R. Zollitzch, Niagara Mohawk Training Supervisor
M. Dooley, Niagara Mohawk NMP2 Training Supervisor
R. W. Gayne, NMP2 Operations
R. I. Brown, General Physics
E. K. Bates, General Physics

6. Summary of NRC comments made at exit interview:

a. The preliminary results of the operating examinations were positive.

b. The generic strengths and deficiencies noted in paragraph 3 of this report were discussed.

c. It was nnted during the exam that many control room forms were not available in the simulator.

d. It was noted that many procedures were not issued or up-to-date.

e. It was noted that the simulator did not perform well during the examinations; specifically, the following deficiencies were discussed.

1. Back panels were not simulated.

2. Many gauges and indicators were not labeled.

3. Simulator fidelity was lacking in several key areas which contributed to confusion among candidates and examiners.

4. Small "glitches" in software occurred which rendered many malfunctions useless for examination purposes. At times, these "glitches" were not pointed out prior to scenario performance, and their occurrence during the exams further added to confusion among candidates and examiners. In addition, defective malfunctions were not identified in the facility provided material which necessitated last minute changes to scenarios.

f. Facility personnel were very cooperative and helpful during the examination process. The simulator instructors/operators performed very well and were extremely helpful to the examiners despite the simulator problems noted ab .

7. Summary of facility comments and commitments made at exit interview:

a. The examination process was conducted in a professional manner, and the written and operating examinations were fair.

b. The facility acknowledged the simulator deficiencies. It was noted that the simulator was to undergo a six week warranty repair and upgrade period immediately following the examinations and that most, if not all, deficiencies would be corrected well in advance of the next cold licensing period scheduled in mid-December 1985. In addition, the facility committed to providing an accurate, up-to-date listing and description of validated simulator malfunctions for NRC review prior to the December examinations. The facility also encouraged the NRC examiners to return prior to the December exams for inspection of the simulator.

Attachments:
1. NRC Resolution of Exam Review Comments and changes noted to be necessary during grading
2. Written Examination and Answer Key (SRO)
3. Written Examination and Answer Key (RO)

Attachment 1

NRC Resolution of Exam Review Comments and Changes Noted to be Necessary During the Grading

Question Number	Comment/Resolution
1.2	It was noted that one part of the answer implied the other part of the answer. This was considered during the grading. However, it should be noted that the correct reactivity effect had to be discussed.
1.3	Facility asked whether an equation would be an acceptable answer in lieu of a verbal explanation. This was acceptable. During grading, it was noted that the answer in the key for Part c. was incorrect. This was corrected.
1.7	Facility noted that the simulator was programmed to show an oscillating ammeter. This was added to the key as acceptable. However, if an answer said that amps would increase above normal, credit was deducted.
1.8	Facility requested clarification of grading if the wrong administrative requirement was assumed. This was added.
2.1	Facility noted that current plant prints do not show the LOCA signal even though it is in the procedure. This was considered during the grading.
2.4	Facility asked that alternate names for setpoints be accepted for this question and others. This comment was accepted, and alternate setpoint names were added to the key for this question and all others where appropriate. These were verified during the review, and not otherwise mentioned in this report.
2.6.b	Automatic relief action added to the answer at facility request.
2.7	Alternate answer of 200 psig added during grading.
2.8.a	Division I DG added during exam review to item (1), and item (2) added during grading. This item (2) addition resulted in higher grades on most exams.
2.11	Alternate answers from other sources added during exam review.
3.6.a	Alternate answers researched during grading and added to the answer key for item (1).

Question Number	Comment/Resolution
3.8	Alternate grading noted depending on how candidate treated his answer.
4.2	Alternate wording suggesting same results noted to be acceptable.
4.4.b	During the grading, it was noted that answers to part (a) should also be acceptable as part (b) answers because of the wording of the question. Noted in the answer key.
4.5.b	Facility noted that temperature and pressure are controlled by two different means. The answer key was expanded during the review to require both answers.
4.5.c	Facility noted that the procedure only required temperatures to be kept below 105 degrees, and that an answer related to the 90 degree limit would be wrong. It was agreed to take off 0.2 points for the 90 degree answer.
4.7.c	During grading, it was found that several candidates gave absolute pressures rather than a relative pressure. Facility was consulted for the correct absolute value. Facility representative noted that simulator and plant values were different, and that a range of answers could be accepted. This range is noted in the answer key.
4.8	Use of vacuum cleaners added during exam review.
5.04.a	The candidates will not be required to know "68 degrees".
5.08	Both sentences in the answer say the same thing, and a candidate will receive full credit for either answer.
6.06	The grader will accept "no response" if the candidate assumes that only one ADS reset button is depressed.
6.08	Added "RRP trip after 23 seconds" for vessel dome pressure.
8.01	Grader will also accept T.S. 3.0.4 if candidate explains that operability is not known.
8.09	Grader will accept T.S. bases 3/4.3.1 as this is a more correct answer.

Attachment 2

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: NINE MILE POINT 1

REACTOR TYPE: BWR-GE5

DATE ADMINISTERED: 85/06/10

EXAMINER: CRESCENZO, F

APPLICANT: MASTER

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE		CATEGORY
25.00	25.00			5.	THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	25.00			6.	PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	25.00			7.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00			8.	ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00	100.00				TOTALS

FINAL GRADE _________________%

All work done on this examination is my own. I have neither given nor received aid.

APPLICANT'S SIGNATURE

Attachment 2

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY:	NINE MILE POINT 2
REACTOR TYPE:	BWR-GE5
DATE ADMINISTERED:	85/06/10
EXAMINER:	CRESCENZO, F
APPLICANT:	MASTER

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

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25.00	25.00	________	________	6.	PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	25.00	________	________	7.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00	________	________	8.	ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00	100.00	________	________		TOTALS

FINAL GRADE ________________%

All work done on this examination is my own. I have neither given nor received aid.

____
APPLICANT'S SIGNATURE

QUESTION 5.01 (3.00)

a. At the beginning of a fuel cycle, control rod density is approximately [illegible] to 12% at equilibrium full power. Approximately one third into the cycle, the control rod density is about 1[illegible] to 16% at equilibrium full power. Why is there a difference? (1.5[illegible])
b. What effect does this increase in control rod density have on the void coefficient of reactivity? Explain your answer. (1.50)

QUESTION 5.02 (2.50)

a. The reactivity worth associated with equilibrium samarium concentration is about 1.0% dk/k, however, unlike xenon, this concentration does not change during normal reactor power level changes. Why is this? (1.50)
b. When would changes in samarium concentration be considered during reactor operation? ([illegible].00)

QUESTION 5.03 (3.00)

Concerning MCPR:
a. Why is the steady state MCPR limit set higher than the transient MCPR limit? (1.00)
b. Why is the 'Kf' factor employed to increase the limit further? ([illegible].00)
c. Why is the transient MCPR limit set at 1.07 instead of 1.0? (1.00)

QUESTION 5.04 ([illegible].00)

What is the definition of the following terms as given in the Tech Spec[illegible]
a. Shutdown Margin ([illegible])
b. [illegible] Control Rod Pattern ([illegible])

QUESTION 5.05 ([illegible].50)

Explain [illegible] the [illegible] "[illegible] reactivity effect" [illegible] at [illegible] reactor [illegible] you would be likely to observe it. ([illegible])

***** END OF CATEGORY 05 CONTINUED ON NEXT PAGE *****

QUESTION 5.06 (2.50)

a. List three post accident sources of hydrogen gas. (1.50)
b. What are the approximate hydrogen concentrations required to create a FLAMMABLE mixture in air? (0.50)
c. What effect can the presence of steam have on the combustibility of an air hydrogen mixture? (0.50)

QUESTION 5.07 (2.50)

During a rapid power increase, [illegible] for rapid power [illegible] the period quickly becomes <80 [illegible] (2.50)

QUESTION 5.08 (1.50)

Why are installed sources needed in the initial core loading? (1.50)

QUESTION 5.09 (2.00)

Using the below pre-scram radial and axial flux profiles, identify where Xenon concentration will cause a higher than normal rod worth after the scram. (2.00)

QUESTION 5.10 [illegible]

The reactor [illegible] Ten (10) hours [illegible] Shutdown Margin [illegible] to the Shutdown [illegible]

NOTE: [illegible]

QUESTION 5.11 (1.00)

TRUE OR FALSE:

By increasing the condenser vacuum (lowering [illegible]) turbine efficiency will decrease but overall cycle efficiency will increase.

(1.00)

***** END OF CATEGORY 05 *****

QUESTION 6.01 ([illegible])

Describe the interlock [illegible] with [illegible] of the refueling platform bridge? ([illegible] required) [illegible]

QUESTION 6.02 (2.00)

Concerning reactor vessel water level indication:
a. Where within the vessel is "instrument zero"? [illegible]
b. What drywell temperature are the normal [illegible] indicators calibrated for? [illegible]
c. Assuming the reactor is pressurized, what problem [illegible] with level indication if drywell temperature [illegible] calibration temperature during normal operation [illegible] (level below lower instrument tap) condition? [illegible]

QUESTION 6.03 ([illegible])

a. Describe the sequence of events during a [illegible] recirculation pump (assume all interlocks [illegible]). ([illegible])
b. What design features within the recirc system prevent starting a recirc pump and immediately injecting cold [flow] water into the vessel? ([illegible])
c. What reactor vessel thermal limit interlocks must be satisfied prior to starting a recirc pump? ([illegible])

QUESTION 6.04 ([illegible])

a. A control rod is selected for motion [illegible] appears in the rod position window of the [illegible] What does this mean? [illegible]
b. What if the same rod were selected [illegible] four rod display [illegible] indicated [illegible]?

***** [illegible] ON [illegible] *****

6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION [illegible]

QUESTION 6.05 (2.25)

a. What conditions will cause the recirc [illegible] flow [illegible] from automatic to manual? (5 required) [illegible]
b. How does the system ensure [illegible] transition [illegible] mode to the manual mode? [illegible]

QUESTION 6.06 (3.00)

a. Assuming valid initiation signal [illegible] the ADS valves once blowdown has commenced? [illegible] required) [illegible]
b. During blowdown the operator depresses the [illegible] describe the response of the ADS system. [illegible]

QUESTION 6.07 (1.00)

List all methods by which an SRV may be opened. [illegible]

QUESTION 6.08 (2.75)

a. What signals will initiate the Redundant [illegible] System? [illegible]
b. Choose one of the signals given in (a) above and [illegible] response of RPCS. Assume the APRMs remain above the [illegible]

QUESTION 6.09 (3.00)

With the plant initially at 100% [illegible]
Briefly describe the response of the Feedwater [illegible] the following input signal failures [illegible]
Include in your answer a discussion of [illegible] point where [illegible]
a. Feedwater flow signal fails low
b. Steam flow signal fails low.
c. RPV level signal fails low.

***** END OF CATEGORY [illegible] *****

QUESTION 6.10 (1.50)

What purpose does the Feedwater Control System Set Point Setdown function serve, and how does it accomplish its purpose? (1.50)

QUESTION 6.11 (3.00)

a. Briefly explain how the SRM's could be used during accident (low level) conditions to provide a crude method for water level indication within the core regions. [illegible]
b. Describe three limitations which might exist when using the SRM's to determine water level after an accident. [illegible]

QUESTION 6.12 (.50)

TRUE OR FALSE:
a. If during remote shutdown operations the system transfer switches are placed in the EMERGENCY mode, associated system indications in the control room will extinguish but control room component control switches will remain enabled. (.25)
b. Transfer of control to the Remote Shutdown Panel will not prevent auto system response to a LOCA condition. (.25)

***** END OF CATEGORY 06 *****

QUESTION 7.01 (1.50)

When must the following dosimetry be worn?
a. Finger TLD's (0.50)
b. Neutron Badge (0.50)
c. Self reading pocket dosimeter (0.50)

QUESTION 7.02 (1.50)

There are four items which negate the use of an extended RWP. List three of these items. (1.50)

QUESTION 7.03 (3.00)

a. Using the attached copies of figures 8 and 5 from procedure N2-EOP-SPL and N2-EOP-SPT, determine the minimum suppression pool water level allowable given the following plant conditions:
   1. RPV pressure = 500 psig
   2. Suppression pool temperature = 175 deg. F.
   NOTE: show all calculations (2.00)
b. If suppression pool water level cannot be maintained above the curve on figure 8, procedure N2-EOP-SPL directs the operator to commence emergency RPV depressurization. What is the basis for this procedural step? (1.00)

QUESTION 7.04 (1.50)

Concerning PCIS, if a containment isolation had occurred due to an actual lo-lo RPV level or hi drywell pressure, what two administrative precautions must be taken prior to resetting or bypassing the isolation signal?

QUESTION 7.05 (2.50)

According to the Site Emergency Plan, the Emergency Director has [illegible] responsibilities and/or authorities that may NOT be delegated to a subordinate during emergency conditions. List these five (5) responsibilities/authorities.

***** CATEGORY 07 CONTINUED ON NEXT PAGE *****

QUESTION 7.06 (2.00)

According to procedure N2-EOP-RL, what precautions must be taken PRIOR TO placing an ECCS system in manual? [illegible]
What precaution must be taken WHILE an ECCS system is in manual? [illegible]

QUESTION 7.07 (2.00)

According to N2-EOP-CPC, SGGT may not be used to control containment pressure when the temperature of the space being evacuated is greater than 212 degrees F. Why is this limitation in effect? (2.00)

QUESTION 7.08 (3.00)

a. During the 'steam condensing mode' of RHS, explain how reactor cooldown rate is controlled. [illegible]
b. State the two reasons which make the RHS loop 'B' of shutdown cooling more desireable than the RHS loop 'A'. [illegible]

QUESTION 7.09 (2.00)

A precaution in N2-OP-92, Neutron Monitoring System, states that 'BWR cores typically operate with neutron flux noise. Care should be taken when operating in this area.'
a. What problem can this noise create? [illegible]
b. In what specific operating condition is this applicable? [illegible]
c. What actions are required if this conditions exists? [illegible]

QUESTION 7.10 (3.00)

a. Procedure N2-EOP-C2, RPV Emergency Depressurization, provides [illegible] of [illegible] methods, in [illegible] order of preference, to reduce [illegible] pressure in the event [illegible] only 3 SRVs can be opened. List these methods in correct order of preference. [illegible]
b. What is the reason for specifying an order of preference? [illegible]

***** CATEGORY 07 CONTINUED ON NEXT PAGE *****

QUESTION 7.11 (1.00)

According to the start-up procedure, how is the SRM [illegible] overlap supposed to be verified on a reactor startup?

QUESTION 7.12 (5.00)

Complete the following chart of Operational conditions.

CONDITION	MODE SWITCH POSITION	AVE. REACTOR COOLANT TEMP.
1. Power Operation	Run	Any Temperature
2. Startup	(a.)	(b.)
3. Hot Shutdown	(c.)	(d.)
4. Cold Shutdown	(e.)	(f.)
5. Refueling	(g.)	(h.)

***** END OF CATEGORY 07 *****

QUESTION 8.01 (2.00)

The plant is operating at 45% power during a startup. A safety relief valve inadvertantly opens and closes. Using the attached Technical Specifications, answer the following questions.

a. What surveillance must be performed concerning the drywell vacuum breakers? (1.00)

b. If two vacuum breakers sets fail the required surveillance, what actions are required? (1.00)

QUESTION 8.02 (3.00)

The plant is at 100% power, and has been for the last 3 days. A computer P-1 edit indicates the need for a TIP trace in all areas of the core. The reactor analyst informs you of an inoperability of 3 of the 5 TIP machines that is not repairable. You are the shift supervisor. What are your actions? (3 actions required plus a brief discussion; reference any sections of the attached T.S. used in your answer). (3.00)

QUESTION 8.03 (3.00)

The reactor is operating in mode 2 at 8% reactor power. A check of conditions/surveillances required to enter mode 1 indicates that one trip system of the ATWS pump trip instrumentation is inoperable. Using the attached Technical specifications, determine if the startup may continue and the mode switch placed in RUN. Site specific references to tech specs for your answer. (3.00)

QUESTION 8.04 (2.00)

The reactor is operating at 100% reactor power and 100% recirculation flow with the two recirculation loop flow control valves in the same position. During a routine surveillance, it is found that one jet pump in the 'B' loop has a diffuser-to-lower plenum differential pressure 20% greater than normal. Using the attached technical specifications, can the plant continue to operate under these conditions? Reference any sections of T.S. used in your answer. (2.00)

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.05 (2.00)

Discuss the actions necessary to make a temporary change to a procedure. (2.00)

QUESTION 8.06 (2.00)

Concerning the placement and restoration of jumpers/blocks or lifting of leads:

a. Who must authorize the installation of jumpers? (0.50)
b. Who must be notified prior to the installation of a jumper? (0.50)
c. Who may be authorized to install or remove jumpers? (3 required) (1.00)

QUESTION 8.07 (2.50)

If a Control Rod Drive is requested for markup and the mode switch is in STARTUP or RUN, what precautions must be taken prior to issuing the markup? (2.50)

QUESTION 8.08 (2.00)

Explain how a loss of normal Reactor Building Ventilation affects secondary containment integrity. (assume no Standby Gas Treatment automatic start) (2.00)

QUESTION 8.09 (2.00)

What is the Tech Spec basis for the selection of the RPS instrumentation setpoints? (2.00)

QUESTION 8.10 (2.00)

When in operational condition 1, the chloride limit of the reactor coolant is to be maintained less than or equal to 0.2 ppm; however, when in conditions 2 or 3, chlorides are to be maintained less than or equal to 0.1 ppm. Why do different limits exist for different operational conditions? (2.00)

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.11 (2.50)

List the shift manning requirements (per tech spec) for the following situations.

a. Shutdown Condition with core alterations in progress. (1.25)

b. Operations greater than 8 hours without process computer. (1.25)

(***** END OF CATEGORY 08 *****)

(************** END OF EXAMINATION ****************)

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Net work out)/(Energy in)

$w = mg$

$s = V_0 t + 1/2\ at^2$

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_0)/t$

$A = \lambda N$

$A = A_0 e^{-\lambda t}$

$PE = mgh$

$V_f = V_0 + at$

$\omega = \theta/t$

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

$W = v\ \Delta P$

$A = \frac{\pi D^2}{4}$

$t_{1/2}eff = \frac{[(t_{1/2})(t_b)]}{[(t_{1/2}) + (t_b)]}$

$\Delta E = 931\ \Delta m$

$\dot{m} = V_{av} A \rho$

$I = I_0 e^{-\Sigma x}$

$\dot{Q} = \dot{m} Cp \Delta t$

$\dot{Q} = UA \Delta T$

$Pwr = W_f \Delta h$

$I = I_0 e^{-\mu x}$

$I = I_0\ 10^{-x/TVL}$

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

$P = P_0 10^{SUR(t)}$

$P = P_0 e^{t/T}$

$SUR = 26.06/T$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$T = (\ell^*/\rho) + [(\beta - \rho)/\bar{\lambda}\rho]$

$T = \ell/(\rho - \beta)$

$T = (\beta - \rho)/(\bar{\lambda}\rho)$

$\rho = (K_{eff} - 1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$\rho = [(\ell^*/(T\ K_{eff})] + [\bar{\beta}_{eff}/(1 + \bar{\lambda}T)]$

$M = 1/(1 - K_{eff}) = CR_1/CR_0$

$M = (1 - K_{eff0})/(1 - K_{eff1})$

$SDM = (1 - K_{eff})/K_{eff}$

$\ell^* = 10^{-4}$ seconds

$\bar{\lambda} = 0.1$ seconds^{-1}

$P = (\Sigma \phi V)/(3 \times 10^{10})$

$\Sigma = \sigma N$

$I_1 d_1 = I_2 d_2$

$I_1 d_1{}^2 = I_2 d_2{}^2$

$R/hr = (0.5\ CE)/d^2$ (meters)

$R/hr = 6\ CE/d^2$ (feet)

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.
1 ft. H_2O = 0.4335 lbf/in.

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf

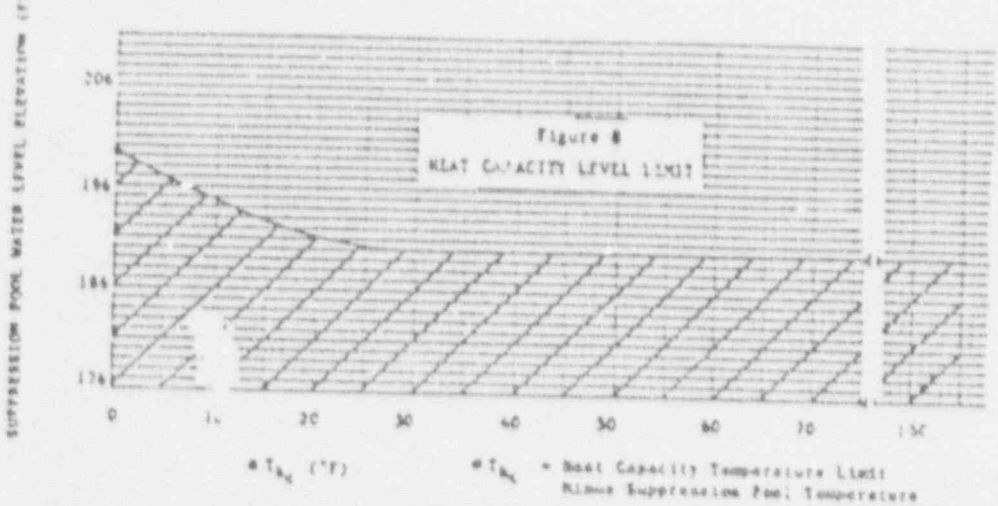


Figure 8
HEAT CAPACITY LEVEL LIMIT

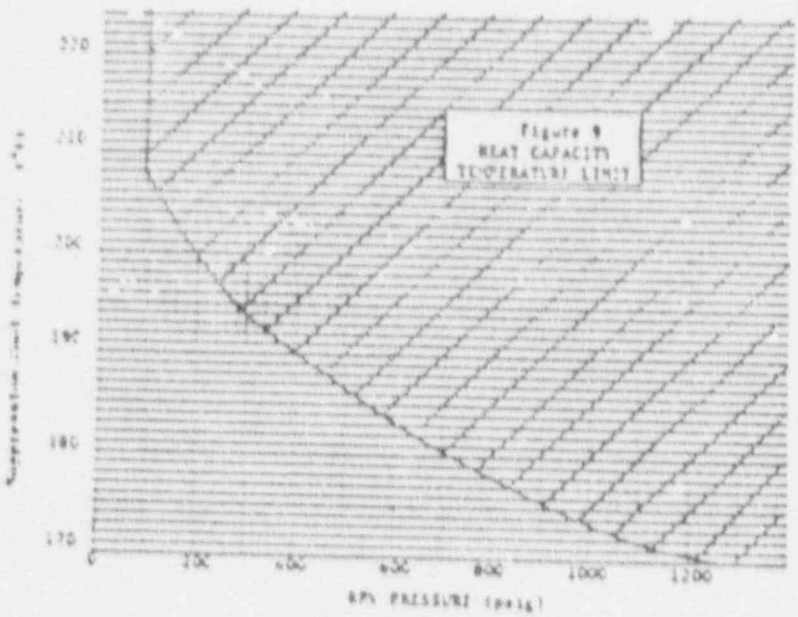


Figure 9
HEAT CAPACITY
TEMPERATURE LIMIT

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Action requirements is not required.

3.0.4 ~~3.0.3~~ When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This specification is not applicable in OPERATIONAL CONDITION 4 or 5.

3.0.5 ~~3.0.4~~ Entry into an OPERATIONAL CONDITION or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

3.0.3 *When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of specification 3.0.4. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in a [illegible] stated in the individual [illegible].*

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specificatons. Surveillance requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	~~At least~~ once per 7 days
Monthly	~~At least~~ once per 31 days
Quarterly or every 3 months	~~At least~~ once per 92 days
Semiannually or every 6 months	~~At least~~ once per 184 days
Every 9 months	~~At least~~ once per 276 days
Yearly or annually	~~At least~~ once per 366 days

SURVEILLANCE REQUIREMENTS (Continued)

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed (13.4) kw/ft.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (25)% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than (25)% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figure 3.2.1-1, ~~3.2.1-2, and 3.2.1-3~~..

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (25)% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, ~~3.2.1-2, or 3.2.1-3,~~ initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than (25)% of RATED THERMAL POWER within the next 4 hours.

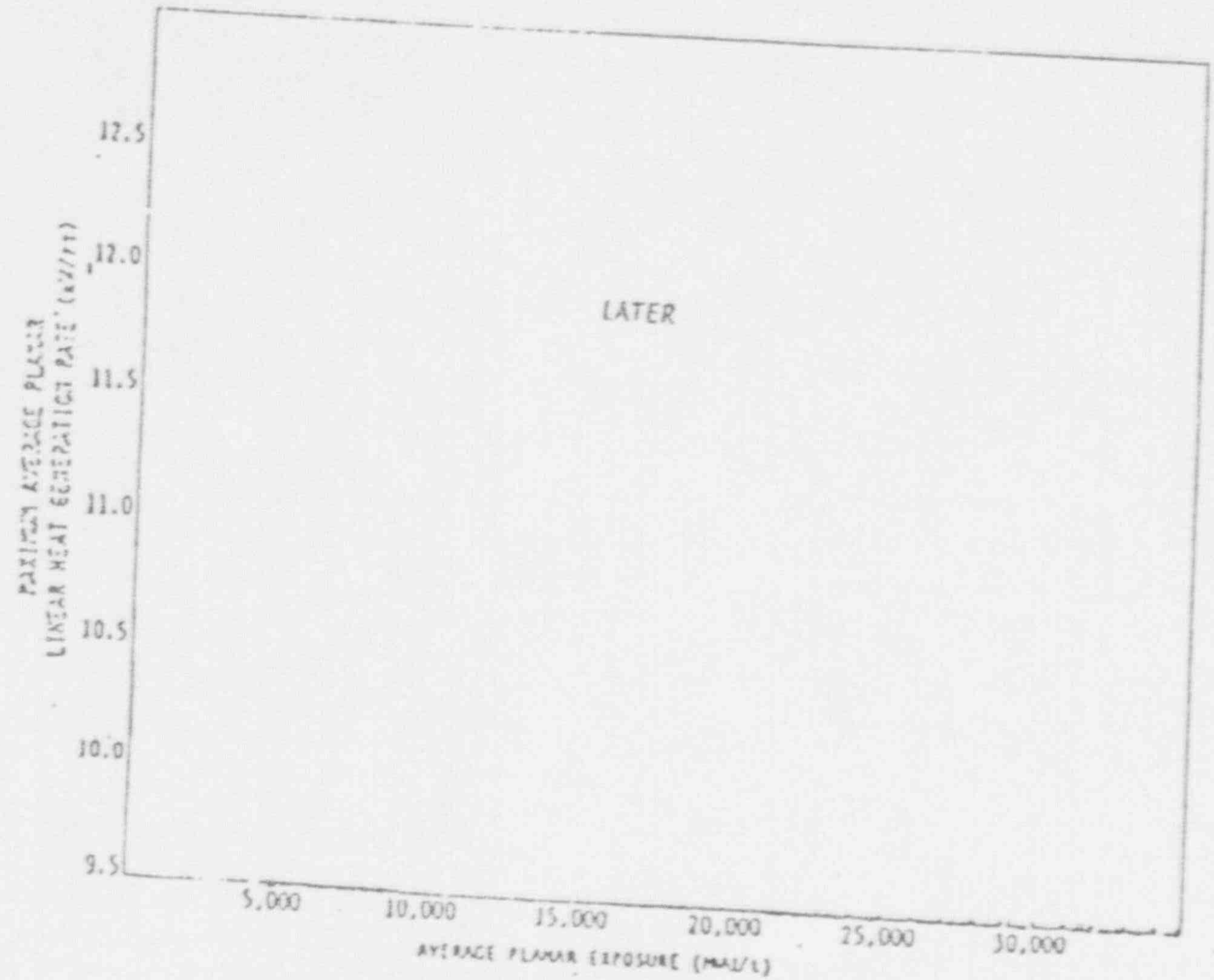
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, ~~3.2.1-2, and 3.2.1-3~~:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPES 8CR183, 8CR233 AND 8CR711

Figure 3.2.1-1

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

TRIP SETPOINT	ALLOWABLE VALUE
$S \leq (0.66W + (51)\%)T$	$S \leq (0.66W + (54)\%)T$
$S_{RB} \leq (0.66W + (42)\%)T$	$S_{RB} \leq (0.66W + (45)\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of ~~(105.0)~~ 108.5 million lbs/hr.
T = Lowest value of the ratio ~~of (design TPF, (2.43) for (8 x 8) fuel, divided by the MTPF obtained for any class of fuel in the core)~~ (FRACTION OF RATED THERMAL POWER divided by the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY). ~~T is always less than or equal to 1.0.~~ T is applied only if less than or equal to 1.0

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (25)% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value(*) within 2 hours or reduce THERMAL POWER to less than (25)% of RATED THERMAL POWER within the next 6 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The ~~(MTPF)~~ (FRTP and the ~~MFLPD) for each class of fuel~~ CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with ~~(MTPF)(MFLPD)~~ CMFLPD greater than or equal to ~~(2.43)~~ (FRTP).

(*With ~~(MTPF)(MFLPD)~~ CMFLPD greater than the ~~(design TPF)~~ (FRTP) ~~during power ascension up to 90% of RATED THERMAL POWER~~, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times ~~(MTPF) (MFLPD)~~ CMFLPD, provided that the adjusted APRM reading does not exceed 100% ~~of RATED THERMAL POWER~~ and a notice of adjustment is posted on the reactor control panel.)

3/4.2.3 MINIMUM CRITICAL POWER RATIO ~~(Optional ODYN Option A)~~

LIMITING CONDITION FOR OPERATION

(1.24)

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit times the K_f ~~shown in from Figure 3.2.3-1, (provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2), with MCPR for 8 x 8 fuel = (1.20)~~.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to [25]% of RATED THERMAL POWER.

ACTION:

~~a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within one hour, MCPR is determined to be greater than or equal to the MCPR limit times the K_f shown in Figure 3.2.3-1, from:~~

~~1. Beginning-of-cycle (BOC) to end-of-cycle (EOC) minus (2000) MWD/t, with MCPR for (8 x 8) fuel = (1.27).~~

~~2. EOC minus (2000) MWD/t to EOC, with MCPR for 8x8 and 8x8R fuel = (1.27).~~

a ~~b.~~ With MCPR less than the MCPR limit times K_f shown in Figure 3.2.3-1, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than [25]% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR shall be determined to be equal to or greater than the MCPR limit determined from Figure 3.2.3-1:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

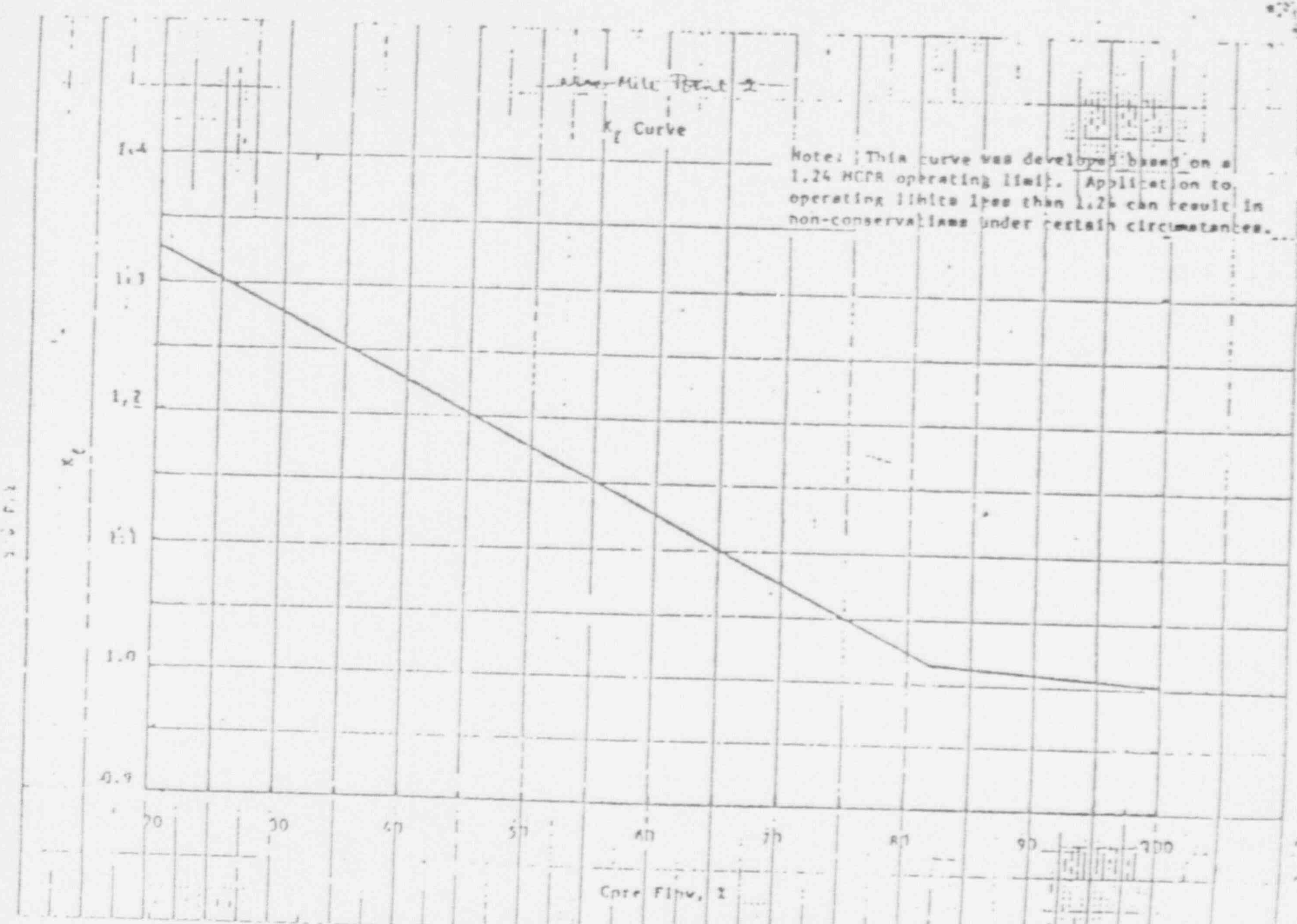

Three Mile Island 2
K_f Curve
Note: This curve was developed based on a 1.24 MCPR operating limit. Application to operating limits less than 1.24 can result in non-conservatisms under certain circumstances.
1.4
1.3
1.2
1.1
1.0
0.9
K_f
20
30
40
50
60
70
80
90
100
Core Flow, %

FIGURE 5.1.3-1

INSTRUMENTATION

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS-RPT system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum Operable Channels per Trip System requirement for one trip system ~~and:~~ *declare the trip system inoperable*

1. ~~If the inoperable channels consist of one reactor vessel water level~~ channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within one hour.

2. If the inoperable channels include two reactor vessel water level channels ~~or two~~ reactor vessel pressure channels, declare ~~the trip system inoperable~~.

c. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1. Each ATWS recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	≥ - 108.8 ~~(38)~~ inches*	≥ - 101.8 ~~( )~~ inches
2. Reactor Vessel Pressure - High	≤ 1050 ~~(1120)~~ psig	≤ 1065 ~~( )~~ psig

See Bases Figures B 3/4 3-1.

INSTRUMENTATION

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

a. In OPERATIONAL CONDITION 2*, three

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least 3 source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

1. CHANNEL CHECK at least once per:

a) 12 hours in CONDITION 2*, and

b) 24 hours in CONDITION 3 or 4.

2. CHANNEL CALIBRATION** at least once per 18 months.

b. Perfu mance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to moving the reactor mode switch from the Shutdown position, if not performed within the previous 7 days, and

2. At least once per 31 days.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 3 cps with the detector fully inserted.

*With IRM's on range 2 or below.
**Neutron detectors may be excluded from CHANNEL CALIBRATION.

INSTRUMENTATION

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.7. The traversing in-core probe system shall be OPERABLE with:

a. Three movable detectors, drives and readout equipment to map the core, and

b. Indexing equipment to allow all three detectors to be calibrated in a common location.

APPLICABILITY: When the traversing in-core probe is used for:

a. Recalibration of the LPRM detectors, and

b.* Monitoring the APLHGR, LHGR, MCPR, or ~~(TPF)~~ (MFLPD).

ACTION:

With the traversing in-core probe system inoperable, suspend use of the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.~~3~~ 4 and 3.0.~~4~~ 5 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use for the above applicable monitoring or calibration functions.

*Only the detector(s) in the required measurement location(s) are required to be OPERABLE.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.10 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.7.10-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instruments less than the Minimum Instruments OPERABLE requirement of Table 3.3.7.10-1:

a) Within 1 hour, establish a FIRE WATCH PATROL to inspect the zone(s) with the inoperable instrument(s) at least once per hour.

b) Restore the minimum number of instruments to OPERABLE status within 14 days or prepare and submit a report in accordance with 6.9.2

c) The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.10.1 Each of the above required fire detection instruments which are accessible during unit operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.* Fire detectors which are not accessible during unit operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST* during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.7.10.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.7.10.3 The non-supervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated OPERABLE at least once per 31 days.

*This does not include detector sensitivity check.

TABLE 3.3.7.10-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION (Illustrative) (**)		MINIMUM INSTRUMENTS OPERABLE * (# of detectors) HEAT	FLAME	SMOKE

*Based on 10% of detectors not operating - no 2 adjacent detectors inoperable.

A. Reactor Bldg./Aux. Bays

		HEAT	FLAME	SMOKE
201 SW	El. 175'-0"			15(16)
202 SW	"			7(7)
203 SW	"			6(6)
204 SW	"			6(6)
205 NZ	"			7(7)
206 SW	"			6(8)
207 SW	"			7(7)
208 SW	"			11(10)
212 SW	"	12(13)		31(34)
213 SW	"	18(20)		32(35)
211 SW	El. 196'-0"			22(22)
214 SW	"			22(22)
221 SW	El. 215'-0"			26(28)
222 SW	"			36(39)
223 SW	"			36(39)
224 SW	"			25(25)
231 SW	El. 240'-0"			28(31)
232 SW	"	5(5)		29(32)
238 SW	"	1(1)		29(32)
239 SW	"			27(29)
243 SW	El. 261'-0"	5(5)		33(36)
245 SW	"	2(2)		34(37)
252 SW	El. 289'-0"	4(4)		36(39)
255 SW	"	4(4)		30(33)
261 NZ	El. 306'-0"	13(14)		
262 NZ	"			24(26)
271 SW	El. 328'-10"			18(19)
272 SW	"			18(19)
273 SW	"			14(15)
274 SW	"			18(19)
281 NZ	El. 353'-10"			[illegible](35)

TABLE 3.3.7.10-1

(Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION (Illustrative) (**)		MINIMUM INSTRUMENTS OPERABLE * (# of detectors) HEAT	FLAME	SMOKE

* based on 10% of detectors not operating - no 2 adjacent detectors inoperable.

B. Control Building (Zones)

Zone	Elevation	HEAT	FLAME	SMOKE
305 NW	El. 214'-0"			4(4)
306 NW	"			9(10)
307 NW	"			1(1)
308 NW	"			1(1)
309 NW	"			5(5)
310 NZ	"			4(4)
311 NZ	"			5(5)
312 NZ	"			9(10)
321 NW	El. 237'-0"			4(4)
322 NW	"			13(14)
323 NW	"			14(15)
324 NW	"			4(4)
325 NW	"			3(3)
326 NW	"			3(3)
327 NW	"			4(4)
331 NW	El. 261'-0"			18(20)
332 NW	"			5(5)
333 XL	"			[illegible]
334 NZ	"			[illegible]
335 NZ	"			4(4)
336 XL	"			[illegible]
337 NW	"			5(5)
338 NZ	"			4(4)
339 NZ	"			1(1)
340 NZ	"			2(2)
341 NZ	"			2(2)
342 XL	"			4(4)
351 NZ	El. 288'-6"			16(17)
352 NW	"			4(4)
353 SG	"	48(48)		24(24)
354 SG	"	48(48)		48(48)
355 NZ	"			(LTR)
356 NZ	"			18(22)
357 XG	"			8(8)
358 XG	"			4(4)
359 NW	"			[illegible]
360 NZ	"			10(11)
362 SG	"	40(40)		20(22)

TABLE 3.3.7.10-1

(Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION (Illustrative) (**)			MINIMUM INSTRUMENTS OPERABLE * (# of detectors) HEAT	FLAME	SMOKE

* based on 10% of detectors not operating - no 2 adjacent detectors inoperable.

			HEAT	FLAME	SMOKE
B. (cont.)	371 NW	El. 306'-0"			4(4)
	372 NZ	"			(LTR)
	373 NZ	"			24(26)
	374 SG	"	40(40)		20(20)
	375 SG	"	40(40)		20(20)
	376 SG	"			[illegible]
	377 NW	"			3(3)
	378 NZ	"			9(9)
	380 NZ	"			18([illegible])
	387 SG	"	35(35)		28(28)
C.	Diesel Generator Building				
	401 NZ	El. 261'-0"			9(9)
	402 SW	"			6(6)
	403 SW	"			6(6)
	404 SW	"			6(6)
D.	Electrical Tunnels				
	301 NW	El. 215'-0"			[illegible]
	302 NW	"			[illegible]
	303 NW	"			3(3)
	304 NW	"			[illegible]
	236 NZ	El. 237'-0"			[illegible]
	237 NZ	"			[illegible]
E.	Service Water pump pits				
	806 NZ	El. 224'-0"			6(6)
	807 NZ	"			6(6)
F.	Fire Pump rooms				
	804 NW	El. 261'-0"			8(8)
	805 NZ	"			2(2)

(**) List all detectors in areas required to ensure the OPERABILITY of safety related equipment and indicate instruments which automatically actuate fire suppression system.

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2 Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when the recirculation loops are operating at the same flow control valve position.

a. The indicated recirculation loop flow differs by more than 10% from the established flow control valve position-loop flow characteristics.

b. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established patterns by more than 10%.

REACTOR COOLANT SYSTEM

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

a. 5% of rated recirculation flow with core flow greater than or equal to 70% of rated core flow.

b. 10% of rated recirculation flow with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the recirculation loop flows different by more than the specified limits, either:

a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

b. Declare the recirculation loop with the lower flow not in operation and take the ACTION require by Specification 3.4.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

LIMITING CONDITION FOR OPERATION

3.6.4.1 Each pair of suppression chamber - drywell vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more vacuum breakers in one pair of suppression chamber - drywell vacuum breakers inoperable for opening but known to be closed, restore the inoperable pair of vacuum breakers to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one suppression chamber - drywell vacuum breaker open, verify the other vacuum breaker in the pair to be closed within 2 hours. Restore the open vacuum breaker to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With ~~(one)~~ (the) position ~~indicator~~ indication of any suppression chamber - drywell vacuum breaker inoperable:

1. Verify the other vacuum breaker in the pair to be closed within 2 hours and at least once per 15 days thereafter, ~~(and)(or)~~

~~(2. Verify the vacuum breaker(s) with the inoperable position indicator to be closed by (conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.5 psi for one hour without makeup within 24 hours and at least once per 15 days thereafter).)~~

2.3. (Otherwise,) be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days and within 2 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

2. At least once per 31 days ~~by verifying (both) (the)~~ observe position indicator(s) OPERABLE ~~by observing expected valve movement~~ during the cycling test.

3. At least once per ~~18 months by~~ operating cycle;

a) Verifying the opening setpoint, from the closed position, to be ~~less than or~~ equal to ~~(0.5)~~ 0.25 psid, and

b) Verifying ~~(both)~~ (the) position indicator(s) OPERABLE by performance of a CHANNEL CALIBRATION.

Answers - SRO

ANSWERS -- NINE MILE POINT 2 -85/06/10-CRESCENZO, F

ANSWER 5.01 (3.00)

a. As the reactor operates during the early part of the cycle, the burnable poison depletes more rapidly than the fuel, therefore, control rods must be inserted to hold the power constant. (1.50)

b. As the control rod density increases, the power producing regions of the core bec e more undermoderated; the moderator to fuel ratio is decreasing. In effect, as total power production has remained constant but the power producing volume has become smaller, the operating volume of the core has become undermoderated. Because of this effect, the void coefficient becomes more negative. (1.50)

REFERENCE
NMPC Operations Technology vol. I page I-12-6

ANSWER 5.02 (2.50)

a. The half life of samarium is greater than 10 e16 years so it can be considered stable. Because of this, removal of samarium is accomplished by neutron absorption only. Since the production and removal of samarium are functions of neutron flux, the term for neutron flux may be cancelled from the equation for equilibrium samarium thus making it non-dependent on flux/power level. (1.50)

b. The greatest changes in samarium concentrations occur during initial startup. (1.00)

REFERENCE
NMPC Operations Technology Vol. I, chapter 15

ANSWER 5.03 (3.00)

a. (The transient MCPR limit is determined by considering uncertainties in monitoring the core operating state) The steady-state MCPR limit considers these uncertainties plus the change in CPR caused by the most limiting transient and thus must be higher. (1.00)

b. The Kf factor is necessary because some transients become more severe at off-design conditions. (1.00)

c. The transient MCPR is set at 1.07 rather than 1.00 to account for the uncertainty in the core operating state. (1.00)

ANSWERS -- NINE MILE POINT 2 -85/06/10-CRESCENZO, F

REFERENCE
NMP II Thermo. Ht. Trans. and Fluid Flow manual chap. 9 pg. 9-93 to 97.

ANSWER 5.04 (2.00)

a. 1.44 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e., (68) degrees; and xenon free. (1.00)

b. 1.23 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR. (1.00)

REFERENCE
NMP II Technical Specifications

ANSWER 5.05 (2.50)

Withdraw. of a control rod causes a decrease in reactor power. (1.00)
This can occur when a shallow rod is withdrawn. As the shallow rod is withdrawn, power increases in the lower portion of the core which creates more voids in the upper portions of the core which tend to overide the positive reactivity of the control rod. (1.50)

REFERENCE
NMP2 LP Control Rod Worth pg 14-11

ANSWER 5.06 (2.50)

a. 1. Zirc-water reactions (0.50)
2. Radiolytic decomposition of water (0.50)
3. Oxidation of vessel and containment components (0.50)
b. 18-59% (0.50)
c. The presence of steam may preclude deflagration or detonation (0.50)

REFERENCE
NMP2 MOCD pg 7-4

ANSWERS -- NINE MILE POINT 2 -85/06/10-CRESCENZO, F

ANSWER 5.07 (2.50)

The period during power increases is governed by how quickly the neutron populat'on can increase. The same holds true on a power decrease however, the neutron population is dominated by the longest lived delayed neutron precursor. This decays with ~80 sec. period. (2.50)

REFERENCE
NMPC Operations Technology pg. I-11-5

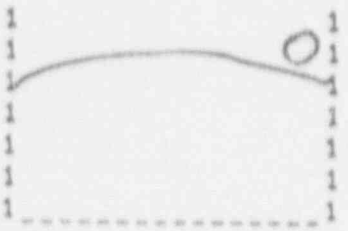
ANSWER 5.08 (1.50)

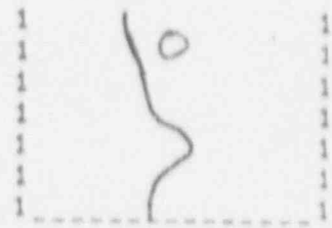
Installed sources are used to raise flux levels in the core to a point where it is on scale for the nuclear instrumentation. Initial intrinsic source levels are not high enough to bring the instrumentation on scale. (1.50)

REFERENCE
NMPC Operations Technology pg. I-5-6

ANSWER 5.09 (2.00)

(2.00)





REFERENCE
NMPC Operations Technology pg. I-14-9

ANSWERS -- NINE MILE POINT 2 -85/06/10-CRESCENZO, F

ANSWER 5.10 (2.50)

With the reactor shutdown by 1% as measured at the time of the peak Xenon, the Shutdown Margin will decrease as Xenon decays. Since peak Xenon reactivity is greater than 1% dk/k, a reactor restart would occur as peak Xenon decays in the next 20 hours. (2.50)

REFERENCE
NMPC Operations Technology Mod. I part 16

ANSWER 5.11 (1.00)

TRUE (1.00)

REFERENCE
NMP Thermo pg 6-76

ANSWER 6.01 (2.00)

Motion block #1 if the following exist: any control rod is withdrawn and either the bridge is near or over the vessel or a load is on one or more of the hoists (1.00)
Motion block #2 if both of the following occur: the reactor mode switch is in 'STARTUP' and the bridge is over the vessel. (1.00)

REFERENCE
NMPC Operations Technology, Fuel Handling and Reactor Servicing Equip. Rev.1, pg. 12

ANSWER 6.02 (2.00)

a. 'Instrument zero' corresponds to the top of the reactor vessel upper grid or 380.69 inches above 'vessel zero'. (0.25)
b. 135 degrees F. (0.25)
c. Indicated water level higher than actual due to decreasing density of water in reference leg or flashing of water inventory in reference leg. Water level can be stable or increasing (on-scale) if actual water level is below the lower instrument tap, concurrent with a loss water inventory in the reference leg due to flashing. (1.50)

REFERENCE
NMPC Operations Technology R.V.I. sys. Rev.1 pg. 16

ANSWER 6.03 (3.00)

a. CB-5 will close to energize the pump windings and start the pump. At the same time CB-1 will close to energize the LFMG. CB-5 will trip when the LFMG is started and the recirc pump is at approx. 95% rated speed. CB-2 will close to energize the pump from the LFMG when the pump speed decreases to 20-26% speed. (1.00)
b. The pump cannot be started unless the flow control valve is in manual and at minimum flow position (interlocked) (1.00)
c. Delta temps between R.V. bottom head drain and steam dome, steam dome and RRS loop suct., and the two RRS loop suctions less than 145, 50, 50, respectively. (1.00)

REFERENCE
NMPC Operations Technology RRS Rev. 1 pg. 9 and10.

ANSWER 6.04 (1.00)

a. A double X indicates that the RPIS is receiving abnormal data (0.50)
b. If a selected rod is a member of a group with less than four rods the display window corresponding to a rod that does not exist remains blank. (0.50)

REFERENCE
NMPC Operations Technology RXMC Rev. 1 pg. 6

ANSWER 6.05 (2.25)

a. 1. Any initiation of high to low recirc pump speed transfer, either auto or manual. (0.25)
2. High drywell pressure. (0.25)
3. Loss of feeds pump concurrent with vessel low level alarm. (0.25)
4. Excessive change in flux controller output signal. (0.25)
5. Deviation of 1% between the loop controller input and manual output signal. (0.25)

b. The flow controllers have a signal tracking unit which automatically matches manual output signal to auto input signal. Also, the controllers shift to manual if deviation exceeds 1%. (1.00)

REFERENCE
NMPC Operations Technology RRFC sys, pgs. 6 and 12

ANSWER 6.06 (3.00)

a. 1. Actuating pnuematics depleted. (0.50)
2. ADS logic is manually reset by the operator. (0.50)
3. Loss of low pressure ECCS pumps. (0.50)

b. The valves will close and the timer will reset and if signals exist the timer will restart. At the end of the timer cycle, 105 sec, blowdown will recommence (1.50)

REFERENCE
NMPC Operations Technology ADS, pg. 7

ANSWER 6.07 (1.00)

1. Manual (0.25)
2. ADS signal (0.25)
3. Spring set pressure (0.25)
4. Pressure control mode (via pressure trip units) (0.25)

REFERENCE
NMPC Operations Technology ADS chapter

ANSWER 6.08 (2.75)

a. 1. High vessel dome pressure. (1050 psig) (0.25)
   2. Low vessel water level. (108.8 in.) (0.25)
   3. Manual initiation (0.25)
b. High pressure: RRP shift to low, start 25 and 98 sec. timer, ARI function, if after 25 sec. the APRMs not downscale then feedpumps runback, after 98 sec. timer SLS initiates if APRMs not downscale.
   Low level: RRP trip, no 25 sec timer
   and RRP Trip
   Manual no RRP shift or trip, no 25 sec. timer (2.00)

REFERENCE
NMPC Operations Technology RRCS pg. 1 table 1

ANSWER 6.09 (3.00)

a. The system sees low feed flow and will increase feedflow until level error balances the feedflow/steamflow mismatch at which point level will stabilize at some new higher level. (1.00)
b. The system sees low steam flow and will decrease feedflow until level error balances the feedflow/steamflow mismatch at which point level will stabilize at some lower level. (1.00)
c. Feed flow will increase and continue to increase until the reactor high level trip. (1.00)

REFERENCE
NMPC Operations Technology FWCS pg 4 and IOP-7 pg. 11 and 12.

ANSWER 6.10 (1.50)

This function prevents overfeeding the reactor following a scram by reducing the level setpoint to one half the normal level selected by the operator. (1.50)

REFERENCE
NMPC Operations T--hnology FWCS pg 4

ANSWER 6.11 (3.00)

a. This could be accomplished by withdrawing the already inserted detectors and observing the count rate. When the detector reaches the water level the count rate should significantly increase due to more moderation of fast neutrons. Knowing the detector withdrawl speed and the time it takes for count rate to increase, the approximate water level can be determined. (1.50)
b. 1. Insertion is to approx. 5 ft. below TAF
   2. May be impossible to detect level if core is completely uncovered.
   3. SRM drive motors may fail under accident conditions.
   4. After 6-8 mos. of shutdown, count rates of a covered core will be low.
   5. Core damage may preclude full or partial detector insertion.
(3 reqd. @ 0.50 ea.)

REFERENCE
NMP 2 MOCD pg. 7-7

ANSWER 6.12 (.50)

a. FALSE
b. TRUE

REFERENCE
NMP 2 LP Remote Shutdown

ANSWERS -- NINE MILE POINT 2 -85/06/10-CRESCENZO, F

ANSWER 7.01 (1.50)

a. Anytime measured or estimated extremity dose rate is more than 6X measured or estimated WB dose rate. (0.50)
b. 1. Neutron dose rate > 2mrem/hr (0.25)
   2. Expected accumulated neutron dose will be > 30 mrem (0.25)
c. 1. Radiation areas (0.25)
   2. High radiatioin areas (0.25)

REFERENCE
RP-1 Access and Radiological Control pg. 13 and 14

ANSWER 7.02 (1.50)

1. Dose rates in an area in excess of 2500 mrad/hr.
2. Contamination levels in excess of 25000 dpm/100cm2
3. Excessive steam or water leakage from contaminated system.
4. Work beyond the scope of the extended RWP. (3 reqd @ 0.5 ea)

REFERENCE
RP-1, Access and Radiological Control page 22

ANSWER 7.03 (3.00)

a. From figure 9, maximum suppression pool temperature for RPV pressure of 500 psig is approximately 187 degrees F. Subtracting 175 from 187 gives a delta temp. of 12 degrees F. From figure 8, minimum suppression pool water level corresponding to 12 degrees F is approximately 194 feet. (2.00)
b. Delta T (heat capacity) is compared to pool level to determine if sufficient water is available to assure complete steam condensation on a RPV depressurization. (1.00)

REFERENCE
NMP LP Suppression Pool Level Control pages 9 and 10

ANSWERS -- NINE MILE POINT 2 -85/06/10-CRESCENZO, F

ANSWER 7.04 (1.50)

1. Health physics must sample the atmosphere in the containment. (0.75)
2. The initiating condition must be corrected. (0.75)

REFERENCE
N2-IOP-83 PCIS pg. 3

ANSWER 7.05 (2.50)

1. Making the decision to notify off-site emergency management agencies.
2. Making protective action recommendations as necessary to off-site emergency management agencies.
3. Classification of the emergency event.
4. Determining the necessity for a site evacuation.
5. Authorizing emergency workers to exceed normal radiation exposure limits. (5 reqd. @ 0.25 ea)

REFERENCE
Site Emergency Plan section 5 pg. 4

ANSWER 7.06 (2.00)

a. Verify misoperation in automatic or adequate core cooling is assured. (1.00)
b. When in the manual mode and auto start is defeated, make frequent checks of initiating parameters. (1.00)

REFERENCE
N2-EOP-RL pg. 3

ANSWER 7.07 (2.00)

Evacuation at temperatures > 212 degrees F can remove non-condesibles and leave a saturated steam environment. Subsequent rapid cooling as by containment spray, may result in pressure reduction to less than design negative differential pressure leading to potential collapse of the space being evacuated. (2.00)

ANSWERS -- NINE MILE POINT 2 -85/06/10-CRESCENZO, F

REFERENCE
N2-EOP-PCP and respective LP

ANSWER 7.08 (3.00)

a. Reactor cooldown rate is controlled by the RHS heat exchanger level. If the level is reduced, more heat exchanger tubes are exposed, and the condensing of reactor steam increases. If the level in the HX is increased, the condensing rate decreases. (1.50)
b. RHS loop B can be flushed and warmed without the use of manual valves and RHS loop B has head spray. (1.50)

REFERENCE
N2-OP-31, Residual Heat Removal sections H.3 and 4

ANSWER 7.09 (2.00)

a. High neutron flux alarm and/or scram. (0.50)
b. At or near 100% rod line, min recirc flow. (0.50)
c. Insert control rods per reactor analyst or increase recirc flow (1.00)

REFERENCE
N2-OP-92 Neutron monitoring, precautions and off normal procedures.

ANSWER 7.10 (3.00)

a. 1. Main Condenser
2. RHR steam condensing mode
3. MS line drains
4. RCIC steam line
5. Head vent (.25 each+.15 each cor. order)

b. The methods are listed in the order which will minimize radioactive release to the environment. (1.00)

ANSWER 7.11 (1.00)

By vissually observing that all IRM's are above the downscale before any SRM count rate is above 10e4 cps with the SRM's fully inserted. (1.00)

ANSWERS -- NINE MILE POINT 2 -85/06/10-CRESCENZO, F

REFERENCE
N2-OP-101A, Plant Startup

ANSWER 7.12 (2.00)

a. Startup/Hot standby
b. Any Temperature
c. Shutdown
d. >200 degrees F
e. Shutdown
f. < or EQ 200 degrees F
g. Shutdown or Refuel
h. < or EQ 140 degrees F (0.25 each)

REFERENCE
NMP LP Reactor Protection System pg. 9 of 32

ANSWER 8.01 (2.00)

a. Cycle each vacuum breaker through at least one cycle of full travel. (1.00)
b. Place the plant in hot shutdown in the nex 12 hours, and in cold shutdown within the following 24 hours. (1.00)

REFERENCE
TS sections 3.6.4.1.a. and 4.6.4.1.b.1

ANSWER 8.02 (3.00)

Since the TIP system provides the computer with the data to calculate thermal hydraulic limits, the assumption must be made that surveillance of APLHGR, LHGR, and MCPR cannot be made. When the required monitoring of these limits cannot be met, the following action statements apply!(1.50)
1. Initiate corrective actions in 15 minutes. (0.50)
2. Restore the limits within 24 hours, or (0.50)
3. Reduce power to < 25% within 4 hours. (0.50)

REFERENCE
T.S. 3.2.1, 3.2.3, 3.2.4.

ANSWER 8.03 (3.00)

Per tech spec section 3.3.4.1.d. the inoperable trip system must be restored to operable status within 72 hours or be in at least startup. The startup may not proceed per tech spec section 3.0.5 which prohibits entry into an operational condition unless the LCO is met without reliance on the action statement. (3.00)

REFERENCE
Technical Specifications sections 3.0.5. and 3.3.4.1.d.

ANSWER 8.04 (2.00)

The jet pump is not inoperable. The surveillance section 4.4.1.2, requires two of the compared parameters (conditions) to be unsatisfactory prior to declaring the jet pump(s) inoperable. Based on this, operation may continue. (2.00)

Technical Specifications section 3.4.1.2, and 4.4.1.2.

ANSWER 8.05 (2.00)

a. The intent of the original procedure is not altered. (0.50)
b. The change is approved by two members of the plant management staff, at least one of whom holds a current SRO license on the unit affected. (0.75)
c. The change is documented, reviewed, and approved by the General Superintendent Nuclear Generation or designee within 14 days of implementation. (0.75)

REFERENCE
NMP-2 Tech Specs, Admin Procedures 6.8.3

ANSWER 8.06 (2.00)

a. Station Shift Supervisor (0.50)
b. Chief Shift Operator (0.50)
c. Licensed operator, qualified I+C technician, electrician (1.00)

REFERENCE
Procedure AP-3.3.2 sect. 2.3

ANSWER 8.07 (2.50)

1. If the control rod is at position '00', the the markup may be issued. (1.00)
2. If the Control Rod is at a position other than '00', then Reactor Physics Group must calculate S.M. for the conditions. If the S.M. requirements are met, then the markup may be issued. If the S.M. requirements cannot be met then the markup may not be issued until the mode switch is in SHUTDOWN or REFUEL. (1.50)

REFERENCE
NMPSO #30

ANSWER 8.08 (2.00)

The normal Reactor Building Ventilation normally maintains the reactor building at -.25 inches W.G. to satisfy the same requirement in the definition of 'Secondary Containment Integrity'. Without this pressure, secondary containment integrity is lost. (2.00)

REFERENCE
NMP2 Technical Specifications, definitions.

ANSWER 8.09 (2.00)

They are selected to ensure that the reactor core and the reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurences and to assist in mitigating the consequences of accidents. (2.00)

REFERENCE
NMP2 Technical Specifications, LSSS Bases

ANSWER 8.10 (2.00)

Chloride limits are specified to prevent stress corrosion cracking of stainless steel. The effect of chloride is not as great when oxygen concentration in the coolant is low thus the higher limit on chlorides is permitted during power operation. (2.00)

REFERENCE
NMPC Operations Technology Reactor Coolant Chemistry pg.5 of 5 App A

ANSWER 8.11 (2.50)

a. 2 SRO's one dedicated to supervising core alterations, 1 RO
1 unlicensed operator. (1.25)
b. 1 SRO, 2RO's, 3 ULO's, 1 ASSS (SRO). (1.25)

REFERENCE
NMP 2 TS Table 6.2-1

QUESTION	VALUE	REFERENCE
05.01	3.00	FJC0000015
05.02	2.50	FJC0000016
05.03	3.00	FJC0000017
05.04	2.00	FJC0000024
05.05	2.50	FJC0000061
05.06	2.50	FJC0000062
05.07	2.50	FJC0000067
05.08	1.50	FJC0000068
05.09	2.00	FJC0000069
05.10	2.50	FJC0000071
05.11	1.00	FJC0000072

	25.00	
06.01	2.00	FJC0000019
06.02	2.00	FJC0000025
06.03	3.00	FJC0000026
06.04	1.00	FJC0000028
06.05	2.25	FJC0000029
06.06	3.00	FJC0000031
06.07	1.00	FJC0000032
06.08	2.75	FJC0000033
06.09	3.00	FJC0000035
06.10	1.50	FJC0000036
06.11	3.00	FJC0000065
06.12	.50	FJC0000066

	25.00	
07.01	1.50	FJC0000038
07.02	1.50	FJC0000039
07.03	3.00	FJC0000040
07.04	1.50	FJC0000041
07.05	2.50	FJC0000042
07.06	2.00	FJC0000043
07.07	2.00	FJC0000044
07.08	3.00	FJC0000045
07.09	2.00	FJC0000047
07.10	3.00	FJC0000060
07.11	1.00	FJC0000073
07.12	2.00	FJC0000074

	25.00	
08.01	2.00	FJC0000048
08.02	3.00	FJC0000050
08.03	3.00	FJC0000051
08.04	2.00	FJC0000052
08.05	2.00	FJC0000054
08.06	2.00	FJC0000056

TEST CROSS REFERENCE

QUESTION	VALUE	REFERENCE
08.07	2.50	FJC0000057
08.08	2.00	FJC0000058
08.09	2.00	FJC0000059
08.10	2.00	FJC0000075
08.11	2.50	FJC0000076

	25.00	

	100.00	

Attachment 3

U.S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

Facility: Nine Mile Point Unit 2

Reactor Type: BWR

Date Administered: June 11, 1985

Examiner: Brian K. Hajek

Candidate: (Print) MASTER

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. ..e passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

Category Value	% of Total	Candidate's Score	% of Category Value	Category
25	25			1. Principles of Nuclear Power Plant Operation, Thermodynamics, Heat Transfer and Fluid Flow
25	25			2. Plant Design Including Safety and Emergency Systems
25	25			3. Instruments and Controls
25	25			4. Procedures - Normal, Abnormal, Emergency, and Radiological Control
100				TOTALS

Final Grade ________ %

All work done on this examination is my own. I have neither given nor received aid.

Candidate's Signature

Reviewers

Ed Bates - GPC - §4

Brian Hennigan - GPC §2

Bob Brown - GPC §1

Jeff §3

1. PRINCIPLES OF NUCL' 'R POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER, AND UID FLOW (25)

1.1 With the reactor operating at 100 percent power, an inadvertent initiation of HPCS occurs. Briefly state how each of the following plant operating parameters will have changed after a new equilibrium operating condition has been reached. If the parameter will have returned to the original or normal condition, state the reason for the return to normal. If the parameter will not have returned to the original or normal condition, briefly explain why the new stable condition or value is present. Assume no operator action is taken. (3.0)

a. RPV pressure
b. Reactor power
c. Vessel level
d. Turbine generator output

1.2 The RRCS employs four methods of minimizing core reactivity should a potential ATWS event occur. Two of these are the ARI function and SLC injection. What are the other two methods. Explain why and/or how each of the latter two methods will reduce core reactivity. (2.5)

1.3 A minimum Net Positive Suction Head (NPSH) is required for assuring that the Recirculation System pumps do not cavitate.

a. Provide a brief definition of NPSH. (0.5)

b. State how the available NPSH changes (increases, decreases, or remains the same) for each of the following:

(1) Reactor water level decreases from normal level to just above the low level scram setpoint. (0.5)

(2) Feedwater heating is lost. (0.5)

(3) Recirc pump speed is changed from low to high frequency. (0.5)

Category Continued on Next Page

1.4 Dissolved oxygen is measured continuously by sampling the condensate as it returns to the reactor.

a. If the dissolved oxygen level is high, how may this adversely affect the operation of the reactor? (1.0)

b. Why will the conductivity of the water increase at the same time the oxygen level is increasing? (1.0)

1.5 Define what is meant by each of the following control rod descriptions, and explain how each of these two types of control rods affect the core flux distribution and/or power when they are moved.

a. Shallow Rods (1.5)
b. Deep Rods. (1.5)

1.6 Procedure IOP-101A, Plant Start-up, warns that extra caution should be used when pulling rods in the region of criticality to avoid short periods. It goes on to say that in previous short period incidents, the operators thought the reactor was substantially sub-critical due to unexpectedly low SRM readings. Explain the reason for this caution and how Xenon Concentration, Moderator Temperature, and the Order of Control Rod Withdrawal can affect the operating condition. (3.0)

1.7 Pump ampere meters are provided in the Control Room for most of the large pumps in the plant. If a system fault occurs that results in a loss of NPSH, such as a manual valve in a suction line having been improperly throttled, describe <u>how</u> and <u>why</u> the meter would behave if the pump began to cavitate. (2.0)

1.8 The reactor is critical at "50" on Range 2 of the IRMs. A control rod is withdrawn three notches, resulting in a power increase with a stable reactor period equal to the minimum permissible sustained positive period permitted in IOP-101A, Plant Start-Up. Heating power is estimated to be at "30" on Range 7. Show all your assumptions and work for the following calculations.

a. What is the doubling time? (1.25)

b. How long will it take to reach heating power? (1.25)

Category Continued on Next Page

1.9 By increasing condenser vacuum (lowering the absolute pressure), turbine efficiency will decrease, but overall cycle efficiency will increase. True or false? (1.0)

1.10 Power plant systems are designed to move water under many different plant conditions over the life of the plant. Pumps are required to be sized to overcome the expected head losses to provide the required flow rates. For each of the following system conditions, indicate whether the head losses will increase, decrease, or remain the same:

a. Scale form on the inside of a pipe (0.5)
b. Pump speed changes from high speed to low speed setting (0.5)
c. Pipe diameter decreases (0.5)
d. A flow control valve is partially closed. (0.5)

1.11 Figure 1.1 shows plots of the coolant temperature and coolant enthalpy as a function of the height in the core or flow path length.

a. Explain why the coolant temperature remains constant over much of the flow path while the enthalpy continues to rise. (1.0)

b. Explain why the curve of coolant enthalpy changes slope over the flow path length. (1.0)

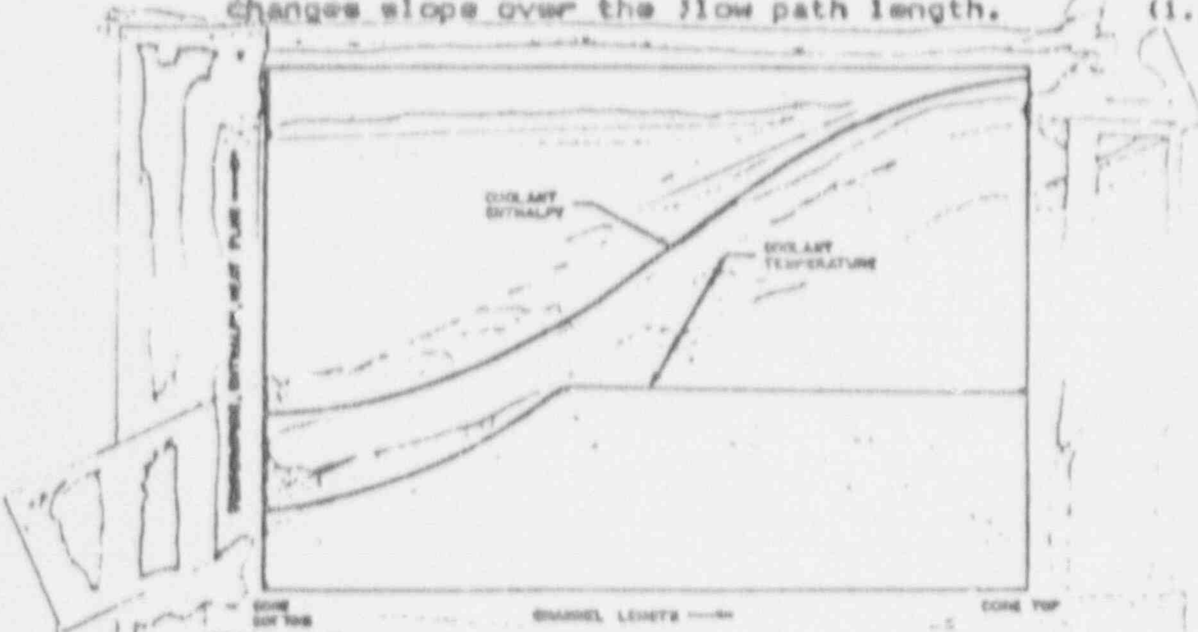


Figure 1.1 Plot of Coolant Temperature and Enthalpy.

End of Category

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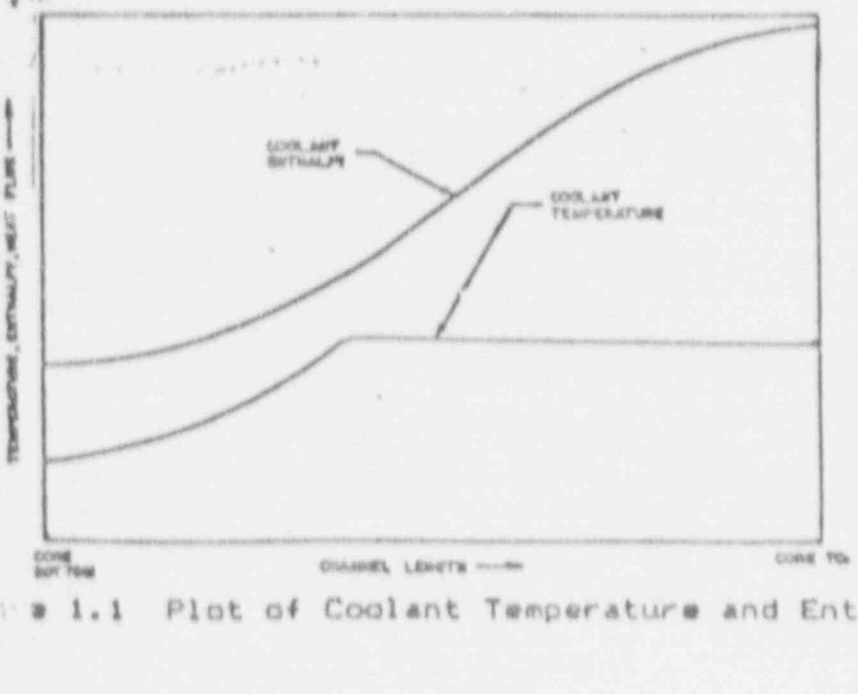

Figure 1.1 Plot of Coolant Temperature and Enthalpy.

End of Category

2. PLANT DESIGN, INCLUDING SAFETY AND EMERGENCY SYSTEMS (25)

2.1 The control room ventilation system supplies tempered recirculated outdoor air to areas on the the 306' elevation.

a. What two conditions (not signals) will cause the standby train to auto start? (1.0)

b. What two conditions will cause an isolation of the normal flow of the control room ventilation system air, and how does the flow path change? (1.5)

c. What action is required prior to operating the control room ventilation system in the smoke clear mode? (0.5)

2.2 In the Control Rod Drive Hydraulic System, Drive Water Header pressure is maintained at 260 psig above reactor pressure. Explain how this pressure is regulated, and how it is maintained when water is required for normal insertion of a control rod. (2.5)

2.3 The reactor recirculation pumps are each provided with a two stage cartridge seal assembly. Water from the Control Rod Drive Hydraulic System is provided to each seal assembly at about 3 to 5 gpm.

a. Normal seal leakage to the Liquid Radwaste System is about one gpm. Where does the rest of the CRD water flow? (0.5)

b. If only the Number 1 seal fails, what will happen to the pressures, flows, and temperatures in each of the two seal cavities? (1.5)

2.4 A small break accident has occurred and the High Pressure Core Spray System has initiated as required. The capacity of HPCS is considerably greater than the leak rate, and the injection is automatically terminated after about ten minutes. (1) What terminated the injection, (2) will the injection automatically restart if needed, and (3) if HPCS does restart (either manually or automatically), what will be the likely source of water? Give setpoints and/or reasons for each of the above three questions. (3.0)

Category Continued on Next Page

2.5 The Reactor Water Cleanup System is designed to enable letdown during a reactor heatup to either the main condenser or to the radwaste collector tank. Explain why it is inadvisable to route all the cleanup flow to the reject system from the filter/demineralizers. Include any operator actions which may be required, or any automatic actions that may occur. (2.0)

2.6 Three solenoid-operated air pilot valves are provided for gas to be admitted into the ADS SRVs pneumatic cylinder actuator.

a. Will ADS initiate if one of these pilot valves fails? Explain your answer. (1.0)

b. All 18 SRVs have these three pilot valves, but only the ADS SRVs use all three. All SRVs use at least one. What is the purpose of the single pilot valve that is used by all 18 SRVs? (0.5)

2.7 Condenser vacuum is established by a mechanical system and maintained by steam jet air ejectors.

a. At approximately what condenser pressure will the steam jet air ejectors be placed in service? (0.5)

b. At approximately what reactor pressure will the auxiliary steam supply be switched from the auxiliary boiler to main steam? (0.5)

c. During normal plant operation, at approximately what pressure must the SJAE driving steam supply pressure be maintained? (0.5)

2.8 The Low Pressure Core Spray System injects water into the Reactor Pressure Vessel in the case of a large break accident.

a. If you need to perform a full flow test of this system, why is it important for you not to initiate the system using the S9 pushbutton manual initiation switch on Panel 601. (1.0)

b. If a valid initiation signal is received while you are performing your test, will the injection valve (MOV104) open immediately? Explain your answer. (1.0)

Category Continued on Next Page

2.9 During Shutdown Cooling Mode of operation of the Residual Heat Removal System, it might be possible to drain the reactor vessel. What interlocks should prevent this from happening? (2.0)

2.10 You have been sent to help troubleshoot a fault that is preventing electrical power to be supplied to Service Water Pump A. You are trying to follow the electrical cables from the switchgear to the pump. You have just entered a cable spreading area with cable trays color coded in purple. Can you continue your task in this area? Why or why not? (1.0)

2.11 Four vacuum relief lines are provided between the Suppression Pool and the Drywell. What is the purpose of these lines and how do they perform their function? (1.5)

2.12 The Drywell Equipment and Floor Drain System collects and provides for excessive leakage detection of water from equipment inside the drywell.

a. The Equipment Drain System includes a cooler. Where is this cooler located in the flowpath, and why is it necessary? (1.0)

b. Two equipment drain pumps take a suction on the drain tank and discharge to the Liquid Radwaste System. Briefly describe the operation of these pumps, including the initiating signals and the sequencing. (2.0)

End of Category

3. INSTRUMENTS AND CONTROLS (25)

3.1 The Safety Parameter Display System provides a top level display and four secondary displays of plant parameters.

a. For the top level display, what are the significances of red and green bar charts? (1.0)

b. If one of the four parameters indicated at the bottom of the various displays is (1) in the "alert" status, or (2) in the "failed" status, how will the display indicate this status to the operator? (2.0)

3.2 Explain how the indicated water level would differ from the actual level (higher than, lower than, or the same as actual) for the following conditions. Include WHY it responds in the way you indicate.

a. Increased Drywell temperature as a result of a small LOCA near a Recirc Pump suction nozzle (1.5)

b. Level instrument reference leg leak. (1.5)

3.3 The status of the squib valves in the Standby Liquid Control System is monitored by light indications on the Control Room panel. What actually is being monitored, how is the monitoring performed, and how is it assurred that the squib valves will not fire inadvertently as a result of the monitoring process? (1.5)

3.4 List six automatic isolation signals that will cause an MSIV closure if the reactor is operating at full power. Include their setpoints. (3.0)

3.5 The Redundant Reactivity Control System initiates four independent actions to prevent and/or to mitigate the potential consequences of an ATWS event.

a. Why do the RRCS-ARI actuation signals seal in for 30 seconds? (0.5)

b. What conditions will result in RRCS initiating SLC? (1.5)

(2.5

Category Continued on Next Page

3.6 The Reactor Recirculation Flow Control System includes three levels of control. While operating in Master Manual control,

a. What five conditions will cause the Loop Flow Controllers to automatically transfer from Automatic to Manual? (2.0)

b. What two conditions will cause the Flux Controller to automatically transfer from Automatic to Manual? (1.0)

3.7 The Instrument and Service Air systems receive air from a common set of three air compressors.

a. The control switches must be in the auto after stop (green flag) position during normal operations. If the standby compressor started, what would the consequences be of matching the flag to the running status? (0.5)

b. If the air header pressure continued to drop after the standby compressor started, and the Service Air System isolated, what action is required to restore Service Air? (0.5)

c. If Instrument Air were to be completely lost, in what position would each of the following valves fail? (2.0)

(1) Scram Inlet Valves
(2) Reactor Water Cleanup Filter/Demin inlet and outlet valves
(3) Cooling Tower Level Control Valve
(4) Condenser 4" make-up valve (LV-103, normal make-up)

3.8 The Nuclear Instrumentation is designed to provide for channel overlap during startup and shutdown operations. Various interlocks are provided that will lead to either rod blocks, reactor scrams, or that preclude operation of detector drives. What are the three SRM System rod blocks, and when are they bypassed as a function of IRM settings or indications? (1.5)

Category Continued on Next Page

3.9 The Reactor Building Closed Loop Cooling Water System provides cooling water to auxiliary equipment and systems.

a. What temperature (where is it sensed) is used to control RBCLC System temperature? (0.5)

b. For the condition of a system temperature higher than normal, describe how the temperature control system will react to bring the temperature back to normal. Include locations of the major components involved in temperature control, and a description of how these components will respond in this situation. (1.5)

3.10 While the reactor is operating at 100 percent power in three element level control, one of the steam flow inputs is lost. Discuss the effect this will have on the reactor level. Be sure to include in your discussion how the various inputs are derived, how error signals are developed and used, and what the final approximate indicated and actual parameter values (reactor power, steam flow, FW flow, and level) will be. (3.0)

End of Category

4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL (25)

4.1 According to IOP-29, Reactor Recirculation System, what are the permissible recirculation loop flow mismatch conditions for two-pump operation? (2.0)

4.2 With the reactor operating at 100 percent and no indicated transient activity occurring, you receive a Control Room alarm indicating that a Service Water pump has automatically started. The reactor continues to operate normally. The pump continues to run, and upon checking, you note that the control switch is in Normal After Start. According to IOP-11, Service Water System, could the pump have started due to a valid start signal? Why or why not? What will need to be checked if this event should occur? (2.0)

Announced to all candidates

4.3 According to IOP-21, Main Turbine, there are several precautions and time limitations associated with turbine startup to assure proper operation, warmup, and to preclude damage from excessive vibration.

a. How long prior to startup shall the turbine be on the turning gear? (0.5)

b. Why should shell warming begin as soon as possible after steam seals are established, and what might result if shell warming is excessively delayed? (1.0)

c. What might occur if first stage pressure exceeds 90 psig during shell warming? (1.0)

4.4 According to AP-4, Administration of Operations,

a. What two items must be signed off by the CSO during a shift change in the Control Room? (1.0)

b. List four items that must be documented as having been reviewed by the oncoming CSO. (2.0)

Category Continued on Next Page

4.5 According to IOP-31, Residual Heat Removal System, one of the normal operational modes is the Steam Condensing Mode.

a. In this mode, steam is condensed and rejected to the Suppression Pool for a while prior to being returned directly to the reactor pressure vessel. Why is the condensate rejected to the Suppression Pool, and when may it be valved over for return to the reactor? (1.0)

b. A precaution states that the shell side of the RHR heat exchangers should not exceed 500 psig or 480 degrees F. How are pressure and temperature controlled to assure that these limits will not be exceeded? (1.0)

c. While operating in the Steam Condensing Mode, it may be necessary to remove one of the two heat exchangers from Steam Condensing to perform another function. What is this other function, and why would it need to be performed? (1.0)

4.6 Entry conditions for the EOPs may be common among several, thus requiring entry into more than one at a time. Consider the condition of falling reactor water level. You have appropriately entered N2-EOP-RL, RPV Water Level Control.

a. What other EOPs must also be entered? (0.5)

b. Why is concurrent execution of these procedures necessary? (1.0)

c. If the low water level condition clears after you have entered N2-EOP-RL and before you have exited the procedure, and then reoccurs, what action relative to procedure execution must you take? (0.5)

Category Continued on Next Page

4.7 During a plant startup and heatup, several actions must be taken as a function of RPV pressure. For each of the following actions, give the approximate pressures by which, at which, or above which, the action must be taken according to IOP-101A, Plant Start-Up.

a. The ADS must be verified operable prior to reactor pressure exceeding ______________. (0.5)

b. Condenser vacuum must be established prior to opening a bypass valve with the vacuum being maintained by the SJAEs. The EHC will open a bypass valve at approximately ______________. (0.5)

c. Start a motor driven feedpump when reactor pressure reaches about ______________ ______________________________________. (0.5)

d. Transfer the Mode switch to RUN after (among verifications of other parameters) the steamline pressure has been verified to be greater than ______________. (0.5)

4.8 What are the six conditions that require a Radiation Work Permit according to RP-2, Radiation Work Permit Procedure? (3.0)

4.9 What are the entry conditions for N2-EOP-SPT, Suppression Pool Temperature Control? Include all setpoints. (2.5)

4.10 IOP-101A, Plant Start-Up, requires that a control rod coupling integrity check be performed for each rod as it is pulled to Position 48, and IOP-96, Reactor Manual Control and Rod Position Indication System, provides instructions for when a rod is not coupled.

a. List the indications you would have if a control rod was found to be uncoupled while you were performing a coupling check. (1.5)

b. If you have indication of an uncoupled control rod, what action are you required to take according to IOP-96. (1.5)

End of Examination

EQUATION SHEET

$f = ma$

$v = s/t$

Cycle efficiency = (Network out)/(Energy in)

$w = mg$

$s = V_o t + 1/2\ at^2$

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

$A = A_o e^{-\lambda t}$

$PE = mgh$

$V_f = V_o + at$

$w = \theta/t$

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

$$t_{1/2}eff = \frac{[(t_{1/2})(t_b)]}{[(t_{1/2}) + (t_b)]}$$

$\Delta E = 931\ \Delta m$

$I = I_o e^{-\Sigma x}$

$\dot{Q} = \dot{m}Cp\Delta t$

$\dot{Q} = UA\Delta T$

$Pwr = W_f \Delta n$

$I = I_o e^{-\mu x}$

$I = I_o\ 10^{-x/TVL}$

$TVL = 1.3/\mu$

$HVL = -0.693/\mu$

$P = P_o 10^{sur(t)}$

$P = P_o e^{t/T}$

$SUR = 26.06/T$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$T = (\ell^*/\rho) + [(\beta - \rho)/\lambda\rho]$

$T = \ell/(\rho - \beta)$

$T = (\beta - \rho)/(\lambda\rho)$

$\rho = (K_{eff} - 1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$M = 1/(1 - K_{eff}) = CR_1/CR_o$

$M = (1 - K_{effo})/(1 - K_{eff1})$

$SDM = (1 - K_{eff})/K_{eff}$

$\ell^* = 10^{-4}$ seconds

$\bar{\lambda} = 0.1\ \text{seconds}^{-1}$

$\rho = [(\ell^*/(T K_{eff})] + [\bar{\beta}_{eff}/(1 + \bar{\lambda}T)]$

$P = (\Sigma\phi V)/(3.12 \times 10^{10})$

$\Sigma = \sigma N$

$I_1 d_1 = I_2 d_2$

$I_1 d_1^2 = I_2 d_2^2$

$R/hr = (0.5\ CE)/d^2(\text{meters})$

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)

1. PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER, AND FLUID FLOW (25) - ANSWERS

1.1 a. The pressure will return to normal [after decreasing slightly]. The EHC system will bring it back to normal.

b. Reactor power will be slightly higher due to the increased subcooling caused by the HPCS flow.

c. The vessel level will be restored to normal by the level control system (reduction in FW).

d. The plant electrical output will be reduced due to the quenching or collapsing of voids above the top of the reactor by the HPCS flow.

0.25 for the final state, and 0.5 for the reason

REFERENCE: Simulator Malfunction Cause and Effects, CS01

1.2 1. Trip the reactor recirc pumps to low speed or to full off. (0.5)

This will cause an increase in core voids which will insert negative reactivity because of the negative void coefficient. (Note that one part of answer implies the other parts) (0.75)

2. Runback the FW control valves to fully closed or zero flow. (0.5)

This will cause a reduction in inlet subcooling. The average moderator temperature at the bottom of the core will increase and cause an insertion of negative reactivity due to the negative moderator temperature coefficient. (0.75)

REFERENCE: NM2 Operations Technology, RRCS, ppg. 4 -5, and Reactivity Coefficients I.

1.3 a. The difference between static pressure [at the eye of the pump] and saturation pressure. $NPSH = P_{ACT} - P_{SAT}$

b. Decreases
c. ~~Decreases~~ Increases
d. Remains the same

REFERENCE: GE T, HT, and FF Text, ppg. 7-91 - 7-96.

1.4 a. Chloride stress corrosion occurs under conditions of high temperatures, high chloride levels, and high oxygen levels. [If the oxygen level is kept low, the Cl level may be higher without causing excessive corrosion rates.] etc

b. High dissolved oxygen concentrations are primarily a result of air leaks. The air also contains carbon dioxide and other gases that form ions in the water.

REFERENCE: NM2 Operations Technology, BWR Chemistry, ppg. 50 - 51.

1.5 a. Shallow rods are those rods that are notch positions 32 - 48. (0.5)
They are also called shaping rods. They affect the local axial flux individually, while taken together, they affect the radial power distribution depending on pattern and symmetry.

b. Deep rods are those rods that are at notch positions 00 - 16. (0.5)
They are also called power rods. Deep rods affect the actual power production in the core.

REFERENCE: NM2 Operations Technology, Module I, Part 14, pg. 10.
Suggested by Exam Bank Q # 74.

1.6 (1) Xenon will reduce the flux in the areas of the core in which its concentration is highest. During normal operations, it will be produced in the areas of highest neutron flux. After shutdown, it will peak most in these areas, and push the flux during startup to areas of lower concentrations. It will also tend to suppress the flux in the areas where the SRMs are located. Thus the overall effect may be that the total average flux level in the core is higher than being indicated by the SRMs. (0.2) (1.5) for total

(2) At higher temperatures, such as you might have at hot standby after a reactor scram, neutrons travel further during slowing down and thus have a greater probability of being absorbed by a control rod. This effectively increases the worth of the control rods. (0.75)

(3) If the rods are withdrawn in a sequence that maintains sufficient radial separation, the flux will be loosely coupled, and the rods may be withdrawn in sequence with decreasing rod worths. That is, the first rod in a group is generally worth the most. (0.75)

REFERENCE: IOP-101A, Precaution D.2.0.

1.7 The motor amps would decrease. (0.5)

Normally, as flow increases through a pump, the pump amperage increases. However, if cavitation occurs in the pump, it takes less power to rotate the impeller in a vapor bubble environment than it does in all water. (Note that the simulator shows oscillation in amps - this is acceptable with correct explanation.) (1.5)

REFERENCE: NM2 Lesson Plan, Fluid Statics, Dynamics, and Delivery, ppg. 16 - 17.

1.8 a. Min permissible stable period = 60 sec* (0.5)

DT = period/1.445 or 1.443

= 60/1.445 = 41.6 sec (0.75)

b. $P = Po\ e^{t/T}$ P/Po = 3000/5 = 600

ln 600 = t/T

t = 60 * ln 600 = 384 sec = 6.4 min

* If the wrong value is given here, candidate will lose 0.5. The candidate's value will then be used as if it were the correct value for grading the rest of Part a and all of Part b

REFERENCE: NM2 Operations Technology, Module 1, Part 10, pg. 2.
IOP-101A, pg. 7.

1.9 True

REFERENCE: BE T, HT, and FF Text, Chapter 6, pg 6-76, Q# 11.

1.10 a. Increase
b. Decrease
c. Increase
d. Increase

REFERENCE: BE T, HT, and FF Text, ppg. 7-40 - 53.
Inspired by EB Q # 46.

1.8 Answers for other values of τ

τ	DT	time
50 sec	34.7 sec	319.9 sec = 5.33 min

1.11 a. The coolant temperature rises continuously as heat (energy) is absorbed until saturation conditions are reached. After this, the temperature remains constant, but the quality increases.

b. Enthalpy changes continuously over the flow path because the water continues to absorb energy from the fuel. The rate of energy absorption is proportional to the heat flux (neutron flux). Since the flux is highest near the center of the core, the slope is greatest at this point.

REFERENCE: NM2 Lesson Plan, Introduction to BWR Thermodynamics and Thermal Hydraulic Limits, pg. 11, and GE T, HT, and FF Text, ppg. 9-31 - 9-33.

End of Category

2. PLANT DESIGN, INCLUDING SAFETY AND EMERGENCY SYSTEMS (25) - ANSWERS

2.1 a. It will auto start on low flow or high temperature of the operating train.

b. 1. High radiation [upstream of the a/c units]
2. LOCA signal
3. The mixture in outside and recirculated air is diverted to the filter units. It is not a true isolation in that outside air is still brought in.

Note that current prints do not include the LOCA sig. The procedure was written from an ECN

c. The normal ventilating a/c units must first be secured.

REFERENCE: IOP-53A, ppg. 1 - 4.

2.2 Drive Water Header pressure is regulated by a manually positioned pressure control valve [PV101]. The flow is throttled from the Control Room. (0.75)

Normally, with no control rod motion, the drive header flow goes through a pair of stabilizing valves. (0.5)

When a control rod is inserted, [4 gpm] is required. (0.5)

To maintain constant flow, one of the two stabilizing valves closes to permit the drive water to be diverted to the control rod mechanism, and to maintain constant flow and pressure drop at the PCV. (0.75)

REFERENCE: IOP-30, ppg. 3 - 4.

2.3 a. It flows through the impeller into the RPV.

b.

Parameter	No. 1 Cavity	No. 2 Cavity
Pressure	No change	Approaches Pressure in No. 1
Flow	No change	Increases Slightly [1.0 - 1.1]
Temperature	Decreases	Decreases

REFERENCE: NM2 Operations Technology, Module 3, Part 6, pg. 3 and Figure 6.

2.4 (1) The injection was terminated when the RPV water level reached 202.3 inches. or Level 8 (0.5)

(2) Yes, HPCS will auto start again when the low Level 2 water level setpoint (108.8 inches) is reached (0.5) or 2 lo

(3) When injection was terminated, the injection valve [107] closed and the min flow valve [105] opened.(0.5) This sends water to the Suppression Pool.(0.3) (0.8)

If the Suppression Pool level increases to 201 feet,(0.4) or if the CST level falls to its low level set point [261 feet or 283,000 gal],(0.4) HPCS should automatically shift its suction to the Suppression Pool.(0.2) For this scenario, the likely source will be the SP because of high level.(0.2) (normal level is 199.6'→201')

(1.0)

REFERENCE: IOP-33, ppg. 1 - 6.

2.5 If all the flow is diverted to the letdown system, no cooling water will be available to the regenerative heat exchanger. (0.5)

This will result in the total heat load being on the non-regen Hx. (0.5)

The non-regen Hx outlet temperature must be kept below 120 F by procedure. If it exceeds this level, it will be necessary to reduce reject flow. (0.5)

If it reaches 140 F, the RWCS will isolate. (0.5)

REFERENCE: IOP-37, RWCS, pg. 22, Section 11.5.

2.6 a. Yes. (0.25)

Two of the valves are provided to auto initiate ADS. If one fails, the other will permit gas to to flow to the actuator,(0.5) since the gas flow line has them in series(0.25). (0.75)

b. It is used for manually opening the SRVs, (0.25)
and for automatic relief action. (0.25)

REFERENCE: NM2 Operations Technology, ADS, ppg. 4 - 7 and Figure 3.
and IOP-34, pg 2.

2.7 a. 23 inches Hg (7 inches Hg absolute)

b. 140 psig < Rx P < 180 psig OR at 200 psig per IOP-101A §2.2

c. 125 psig

REFERENCE: IOP-9, ppg. 3 - 4.

2.8 a. (1) Initiating LPCS with this switch also sends a signal to the RHR system to initiate LPCI with the A RHR pump [and the Division I diesel]. (0.5)
(2) S9 acts just like an auto init and will close the test valve. (0.5)

b. MOV104 won't open until until the differential pressure across the valve is below the open-permissive set point. [700-725 psid]

REFERENCE: Part a. IOP-32, Section E.3.0
Part b. Lesson Plan, LPCS, pg. 9.

2.9 The SDC suction valves (MOV-2A and B) are interlocked to not open if any one of the following are not fully closed: (0.5)

1. Suppression Pool suction valves (1A and B) (0.5)
2. Test Return valves (FV-38A and B) (0.5)
3. Suppression Pool Spray Valves (33A and B) (0.5)

(Valve numbers not required.)

REFERENCE: NM2 Operations Technology, Module 4, Part 5, pg. 9.

2.10 No. (0.25)
All A pumps are in Division I, which is color coded green. (0.5)
Purple is the designation for Division III. (0.25)
[DIV II is yellow]

REFERENCE: NM2 Operations Technology, Module 11, Part 2, pg. 1.

2.11 1. They prevent drawing water from the SP up into the downcomer pipes.
2. They provide drywell floor relief protection during vessel reflood after a LOCA.
3. This is accomplished by limiting the negative pressure differential to 3 psid. } they open to relieve the pressure

REFERENCE: NM2 Operations Technology, Module 58, Part 1, pg. 6.

See also Tech Specs, pg B 3/4 6-5
To maintain structural integrity of the primary containment

(3) ② EOP Lesson Plan Supp. Pool Level Control pg 15 + 16
Redistributing non-condensibles

① NM2 Ops Tech, Mod V, Pt 1, pg 3 re Downcomer pipes for uncondensed steam } NOT VACUUM BKRS

Will accept any 2 of 3 for full credit plus the method

2.12 a. The cooler is located in the flow path before the water enters the drain tank. (0.5)
It is needed to cool the liquid to keep it from flashing to steam when it enters the tank. (0.5)

b. A lead lag control circuit switch automatically selects one pump as lead and one pump as lag. Every time the lead pump starts, the lead-lag selection is switched to assure equal running times. (1.0)
A level switch will start the lead pump. If the level continues to rise, a hi-hi level switch will start the lag pump. When the tank level drops to a low level, a third switch actuates to stop the pump(s). (1.0)

REFERENCE: IOP-67, ppg. 1 - 4.

End of Category

3. INSTRUMENTS AND CONTROLS (25) - ANSWERS

3.1 a. Green indicates normal

Red indicates in alarm or unknown

b. (1) When in the "in alert" status, the numeric value will appear in white followed by a yellow "Q".

(2) When in the "failed" status, the numeric display will be blank followed by an "N" (no color specified).

REFERENCE: NMP Operations Technology, ppg. 3 - 5.

3.2 a. An increased drywell temperature will cause the density of the water in the reference leg to decrease as it heats up. This will result in a lower d/P between the reference and variable legs, and will cause the indicated level to increase, or be higher than actual.

b. A leak in the reference leg will also lower the pressure in this leg, and will cause a lower d/P between the reference and variable legs. The indicated level will be higher than the actual level.

REFERENCE: NM2 Operations Technology, Module 3, Part 3, ppg. 15 - 16.
EB Question # 5.

3.3 The circuit continuity through each firing squib is monitored. (0.5)
A low current [0.01 amps] is passed through the firing circuit. (0.5)
The current is kept considerably below the maximum no fire current rating [of 0.15 amps] to assure the squibs do not fire. [The firing current is 2 amps.] (0.5)

REFERENCE: IOP-36, ppg. 1 - 2.

3.4 1. MSL low pressure 765 psig
2. MSL Tunnel Hi Temp 140 F
3. MSL Tunnel Hi Differential Temperature 50 F
4. Turbine Building MSL Area High Temperature 140 F
5. Main Condenser Low Vacuum 8.5 inches
6. Rx Water Level Lo-Lo-Lo 17.8 inches or Level 1
7. MSL High Radiation 3 x normal
8. MSL High Steam Flow 140 % or 103 psid across an MSIV

Credit for any six.

REFERENCE: NM2 Operations Technology, Module 5, Part 3, Table 1.

3.5 a. To permit full insertion of the control rods. [Venting takes up to 15 sec, and the rods should be in within 25 sec.]

b. 1 High steam dome pressure or 1050 psig and high APRM or 4% (0.3)
or (0.1)
2 low-low water level or level 2 or 108.8" and high APRM (4%) (0.3)
or (0.1)
3 manual initiation signal and high APRM (0.3)
and (0.1)
98 sec time delay timed out in all 3 cases (0.3)
APRM permissive if APRM is inop or above the downscale trip
REFERENCE: NM2 Operations Technology, RRCS, ppg. 6 - 7, and Figure 1.

3.6 a. 1. Any initiation of high to low recirc pump speed transfer. *
2. High Drywell pressure [1.69 psig or 1.68 psig]
3. Loss of a feedpump with concurrent vessel low water level [178.3"] or level 3
4. Excessive rate of change of the Flux Controller output
5. Deviation of 1 percent between the Loop Controller input and manual output signal or Tracking failure of the loop controller

b. 1. Excessive rate of change of the Master Controller output
2. Either Loop Controller transferring to manual

REFERENCE: NM2 Lesson Plan for RRFCS, ppg. 23 - 24.
and L.P. for RRS, pg 12 and IOP-29, pg 4

* 1a ΔT - dome stm to recirc loop suction < 10.7°F for ≥15 sec
b Tot Feed Flow < 30% and FCV < 19% for 15 secs
c Low water level - 159.3" Level 3
d RPT ATWS high dome pressure signal present - 1050 psig
e RPT signal present - 108.8" - Level 2
f. Pwr > 30% and stop vlvs < 90% open or TCV fast closure

REF for

3.7 a. The compressor would not shut down, and if it did shut down, it would not automatically restart.

b. The isolation valve [AOV-171] must be locally reopened. This is done by placing a local switch to open. [It will spring return to normal. If the air header pressure is greater than 85 psig, the valve will open.]

c. (1) Open
(2) Shut
(3) Shut
(4) Open

REFERENCE: IOP-19, ppg. 7 - 10.

3.8

(1) SRM Downscale	IRMs on Range 3 or above
(2) SRM Upscale or INOP	IRMs on Range 8 or above ← If INOP treated separately, then credit given for any 3 of 4
(3) SRM Detector not full in	IRMs on Range 3 or above

REFERENCE: NM2 Operations Technology, Module 6, Part 1, ppg. 15 - 16.

3.9 a. [2CCP-TE108] Hx discharge header temperature or booster pump suction temperature.

b. The operator for a control valve [TV108] receives a signal from the TE. The operator is connected to a tee linkage which is connected to two butterfly valves. One valve controls flow in the Hx bypass line while the other valve controls flow in the common Hx discharge header. As one valve opens, the other closes and vice versa. (1.0)

For this situation, the bypass line valve will close, and the discharge header valve will open. (0.5)

REFERENCE: IOP-13, pg. 1.

3.10 (1) (a) The steam flow signals from the four main steam lines are summed, and (b) the FW flows from two parallel FW lines are summed. (c) With loss of one of the four steam signals, a 25 percent mismatch will exist. (0.6)

(2) (a) This mismatch will produce an error signal which (b) will reduce feed flow to correct the error signal. (0.4)

(3) (a) Since the actual steam flow has not been changed, (b) the level will decrease, and (c) a second error signal will be generated. (d) Since the system is level dominant, (e) a signal will be sent to open the FCVs and increase feed flow to maintain level constant. (1.0)

(4) The final indications will be: (1.0)

Reactor power: 100 %
Steam flow: 100 % actual, 75 % indicated
FW flow: 100 %
Level: Constant, but lower than prior to the fault [A 100 % loss of steam signal is worth about a 7 inch level change, so a one-quarter loss will result in about a 2 inch change.]

REFERENCE: NM2 Operations Technology, Module 9, Part 6, ppg. 4 - 5 and Figure 1.

End of Category

4. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY, AND RADIOLOGICAL CONTROL (25) - ANSWERS

4.1 (1) 5 percent of rated recirc flow with core flow greater than or equal to 70 percent of rated core flow

(2) 10 percent of rated recirc flow with core flow less than 70 percent of rated core flow

REFERENCE: IOP-29, pg. 3, Precaution 3.0.

4.2 No. (0.5)
The only valid auto start signal for Service Water is for load sequencing. (in words leading up to this - s.a. no LOCA) (0.5)
It will be necessary to check the load sequencing signal, and proper continued operation of the rest of the SW System. (1.0)

REFERENCE: IOP-11, I.4.0

4.3 a. At least four hours.

b. This is necessary to prevent uneven heating of the rotor. If it is not started, a rotor long condition could result.

c. The setpoint of the "turbine Stop and Control Valve Closure Bypassed" annunciator could be exceeded, (0.5)
and a reactor scram would result. (0.5)

REFERENCE: IOP-21, Precautions 1, 2, & 3.

4.4 a. (1) The CSO's Control Room Logbook
(2) The shift turnover check sheet

b. (1) Status of tests or special operations in progress.
(2) Special valve/switch lineups.
Mark ways OK
(3) Status and location of jumpers and blocks.
(4) Acknowledgement of instructions from the Operations Supervisor.
(5) Equipment status as reflected in the Equipment Status Log.
(6) A tabulation of key data or special items as required by the Operations Supervisor.

Credit for any four at 0.5 each. — Wording of question requires that if part(a) answers are repeated in(b), they are acceptable.

REFERENCE: AP-4, Section 4.6.

4.5 a. Because Suppression Pool water is used in the RHR system and is normally in the lines, it must be flushed from the system first. (0.5)
It must be flushed from the system until conductivity falls within acceptable limits. H.4.28 (0.5)

b. Temperature is controlled By controlling the level of water on the shell side to expose more or less of the tubes to the steam. [The level should never be lower than 80 percent.] (0.5)
Pressure is controlled by pressure control valves. (0.5)

c. Suppression Pool Cooling. (0.5)
This may be necessary to keep the SP temperature within limits. ~~below~~ [105 F.] (0.2 off for 90°) (0.5)

we water on tube side

105 for testing
110
120

REFERENCE: IOP-31, Sections B.4, D.2, H.4.
NM2 Operation Technology, RHR, pg 7.95 and Figure 1

4.6 a. (1) N2-EOP-RP RPV Pressure Control
(2) N2-EOP-RQ RPV Reactivity Control

b. (1) The actions taken to control any one RPV parameter may directly affect another.

(2) The symptomatic procedures are based on the assumption that the initiating event is not known, and assignment of priorities to any one of the three is not possible.

c. You must return to the beginning of the procedure.

REFERENCE: Lesson Plan, RPV Water Level Control, ppg. 4 - 8.

4.7 a. 100 psig

b. 150 psig

c. 200 psig below the condensate booster pump discharge pressure.

d. 765 psig

REFERENCE: IOP-101A, Sections E. 2.24, 2.30, and 3.5.

Condensate booster pump disch pressure is 550 lbs in the plant but it is simulated @ 700 psig.
∴ Accept answers of 350 lbs → 500 lbs

per John Kaminsky telecon 7/9/85

4.8 (1) Contamination > 10,000 dpm per 100 square cm
(2) Airborne requiring a respirator
(3) Neutron radiation exposure
(4) High Rad Area entries
(5) Unknown conditions
(6) Maintenance in rad or high rad areas.
(7) Use of vacuum cleaners or portable HEPA units in any location in the restricted area
[Using the acronym CAN HUM.]

REFERENCE: RP-2, Section ~~J.1~~.2.1, Jan 85 revision, effective 4/8/85
Accept any six for full credit

4.9 (1) SP Temperature > 90 F
(2) Drywell Temperature > 150 F
(3) SP water level > El. 201 ft.
(4) SP water level < El. 199.5 ft
(5) DW ~~Temperature~~ Pressure > 1.68 psig

REFERENCE: N2-EOP-SPT

4.10 a. (1) Window alarm - "Control Rod Overtravel"
(2) The rod loses the red back light and position indication on the four rod display.
(3) No stall flow is indicated.

b. (1) Verify the auto station response.
(2) Select the rod and drive it in to 44.
(3) Withdraw it again to see if the overtravel condition clears.
Main part of answer.
(4) If it doesn't clear, check Tech Specs for further action.

REFERENCE: IOP-101A, Section E.2.6
IOP-96, Sections D.2 (Precaution), H.4, and I.6.

End of Examination

UNITED STATES
UCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

1 LD

JAN 20 1987

MEMORANDUM FOR: James M. Taylor, Director, IE

FROM: Thomas E. Murley, Regional Administrator, RI

SUBJECT: PROPOSED ENFORCEMENT ACTION - NINE MILE POINT, UNIT 2

Enclosed for your review and concurrence is a proposed enforcement action (letter and Notice of Violation) for three violations which occurred at Nine Mile Point, Unit 2, including two violations of the licensee's technical specification limiting conditions for operation (LCO). Both LCO violations were identified by the licensee, promptly reported to the NRC, and promptly evaluated and corrected. The third violation involved the failure to inform the NRC of a reactor scram in accordance with 10 CFR 50.72.

The first LCO violation involved an inoperable Source Range Monitor (SRM) for approximately five hours during initial fuel load of the reactor. The SRM was inoperable in that its scram function was bypassed during the performance of a surveillance test involving SRM functional tests, but was not returned to service following completion of the test. The ability of the SRM to provide count rate indication in the control room was unaffected. During the time the SRM's scram function was inoperable, 19 fuel bundles were loaded into the reactor in the quadrant in which the SRM was inoperable. This condition existed until identified during a routine control panel walkdown conducted during the first shift turnover following the surveillance test.

[redacted] 1) the scram function for the SRM in an adjacent quadrant was operable; (2) in accordance with station procedures, operators, in direct communication with the refueling floor, were monitoring the count rate provided by the bypassed SRM, and were maintaining an inverse count rate plot; (3) there was no withdrawal of control rods during this time, and only one control rod could have been withdrawn since the mode switch was locked in the refuel position; (4) the scram functions of the Intermediate Range Monitors were operable; and (5) subsequent startup testing verified that both partial and full core shutdown margins were adequate.

The second LCO violation involved the bypassing of all four SRM downscale rod block channels for approximately 2½ hours while the reactor was in the refueling mode. There was no movement of fuel during this time. The four SRM rod block channels were bypassed by installing jumpers so that the reactor mode switch interlock surveillance test could be performed. This condition was contrary to the technical specifications which required that at least two rod block channels be operable. This violation is also considered to be of low safety significance because there was no movement of fuel or control rods during this time.

(D/1)

Information in this record was deleted
in accordance with the Freedom of Information
Act, exemptions 5
FOIA 91-A-2 (90-269) B-3

B-3

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[redacted]

Please note that this memo and Enclosure 1 are being sent on this date to you, the Director of Enforcement, IE, and OGC via the 5520. Enclosure 2, the inspection report, was issued on December 17, 1986, and was previously sent to the Director of Enforcement, IE, and OGC via the Document Control Room.

Thomas E. Murley
Regional Administrator

Enclosures:
1. Letter and Notice of Violation
2. Inspection Report No. 50-410/86-56
3. Licensee Event Report 86-02
4. Licensee Event Report 86-05

cc w/encl:
Enforcement Directors, RII - RIII
Enforcement Officers, RIV - V
B. Beach, IE
J. Lieberman, OGC
K. Abraham, RAD

Releas

Docket No. 50-410
License No. NPF-54
EA 87-

Niagara Mohawk Power Corporation
ATTN: Mr. C. V. Mangan
Senior Vice President
301 Plainfield Road
Syracuse, New York 13212

Gentlemen:

Subject: NOTICE OF VIOLATION
(Inspection No. 50-410/86-56)

This refers to the NRC resident safety inspection conducted between October 1 and November 16, 1986, at Nine Mile Point, Unit 2, Scriba, New York. The inspection report was sent to you on December 17, 1986. During the inspection, NRC inspectors reviewed the circumstances associated with two violations of technical specification limiting conditions for operation (LCO) which were identified by members of your staff and reported to the NRC. Further, a violation of the reporting requirements set forth in 10 CFR 50.72 was also identified. On January 8, 1987, an enforcement conference was conducted with you and members of your staff to discuss the violations, their causes, and your corrective actions.

The two LCO violations occurred during the initial loading of fuel in the reactor. The first violation involved a Source Range Monitor being inoperable for approximately 5 hours during which time 19 fuel bundles were loaded into the reactor. The violation was caused by a lack of independent verification to assure that the SRM was returned to an operable status upon completion of a surveillance test. The second violation involved all four SRM downscale rod block channels being inoperable for a period of approximately 2½ hours while the reactor was in the refueling mode (there was no fuel movement during that time). The violation was caused by an apparent lack of thorough understanding and implementation of technical specification requirements. The violations demonstrate the need for additional training and improved oversight of the operations staff to assure that the plant is operated in accordance with the technical specifications. Therefore, although the two LCO violations are of low safety significance and have been classified at Severity Level IV in accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions," 10 CFR 2, Appendix C (Enforcement Policy) (1986), we emphasize that any similar violations in the future may result in additional enforcement action. The reporting violation has been classified at Severity Level V.

You are required to respond to the enclosed Notice and, in preparing your response, you should follow the instructions specified in the Notice. In your response, you should document the actions taken or planned to prevent recurrence. After reviewing your response to the Notice, including your proposed corrective actions and the results of future inspections, the NRC will determine whether further enforcement action is necessary to ensure compliance with NRC regulatory requirements.

In accordance with 10 CFR 2.790, a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

The responses directed by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget, otherwise required by the Paperwork Reduction Act of 1980, PL 96-511.

Sincerely,

William F. Kane, Director
Division of Reactor Projects

Enclosure:
Notice of Violation

cc w/encl:
Connor & Wetterhahn
John W. Keib, Esquire
J. A. Perry, Vice President, Quality Assurance
W. Hansen, Manager of Quality Assurance
D. Quamme, NMP-2 Project Director
C. Beckham, NMPC QA Manager
T. J. Perkins, General Superintendent
R. B. Abbott, Station Superintendent
Department of Public Service, State of New York
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
State of New York

bcc w/encl:
Region I Docket Room (with concurrences)
Management Assistant, DRMA (w/o encl)
DRP Section Chief
Region I SLO
Robert J. Bores, DRSS

[redacted]

w/h 5

Relea

NOTICE OF VIOLATION

Niagara Mohawk Power Corporation
Nine Mile Point, Unit 2

License No. 50-410
License No. NPF-54
EA 87-

During an NRC inspection conducted between October 1 and November 16, 1986, NRC inspectors reviewed the circumstances associated with two violations of technical specification limiting conditions for operation (LCO) which were identified by the licensee and reported to the NRC. Further, a third violation was identified by the inspector. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (Enforcement Policy) (1986), the violations are set forth below:

A. Technical Specification LCO 3.9.2 requires that whenever the reactor is in the refueling mode, at least two Source Range Monitor (SRM) channels shall be operable and inserted to the normal operating level with one of the required SRM detectors located in the quadrant where core alterations are being performed and the other required SRM detector located in an adjacent quadrant.

   Contrary to the above, between 7:08 p.m. and 11:58 p.m. on November 7, 1986, while the reactor was in the refueling mode, core alterations were made in the C quadrant involving the loading of 19 new fuel bundles into the reactor, and during that time, the SRM channel for the C quadrant was inoperable in that its reactor scram function had been bypassed.

   This is a Severity Level IV Violation (Supplement I).

B. Technical Specification Limiting Condition for Operation (LCO) 3.3.6.b and Table 3.3.6-1 require that whenever the reactor is in the refueling mode, a minimum of two Source Range Monitor (SRM) downscale Rod Block Monitor (RBM) channels per trip function shall be operable. If none of these channels are operable, the Technical Specification LCO Action Statement requires that at least one of the inoperable channels must be placed in the tripped condition within one hour.

   Contrary to the above, between 1:10 p.m. and 3:35 p.m. on November 4, 1986, while the reactor was in the refueling mode, all four SRM downscale rod block channels were inoperable, and during that time, none of these channels were placed in the tripped condition.

   This is a Severity Level IV Violation (Supplement I).

C. 10 CFR 50.72(b)(2)(ii) requires that the NRC be notified as soon as practical and in all cases, within four hours, of the occurrence of any event that results in the unplanned automatic actuation of the Reactor Protection System.

Contrary to the above, at 12:42 a.m. on November 5, 1986, a high water level in the scram discharge volume tank actuated the Reactor Protection System causing a reactor scram, and the NRC was not notified of this unplanned actuation.

This is a Severity Level V violation. (Supplement I)

Pursuant to the provisions of 10 CFR 2.201, Niagara Mohawk Power Corporation is hereby required to submit to this office within thirty days of the date of the letter which transmitted this Notice, a written statement or explanation in reply, including for each alleged violation: (1) admission or denial of the violation; (2) the corrective steps which have been taken and the results achieved; (3) corrective steps which will be taken to avoid further violations; and (4) the date when full compliance will be achieved. Where good cause is shown, consideration will be given to extending this response time.

Dated at King of Prussia, Pennsylvania
this ____ day of January 1987

(611)

POST INSPECTION SALP EVALUATION

FACILITY Nine Mile #2 LICENSEE [illegible] INSP. REPORT NO. [illegible] INSPECTION DATE(s) 5-21-24, 1984 [illegible]

INSPECTOR [illegible] REVIEWED BY ____________ DATE ________

INSTRUCTIONS:

(1) Indicate the functional area inspected [illegible] (operations, refueling, maintenance, etc.). If the inspection covered more than one functional area, complete another Post Inspection SALP Evaluation for that area.

(2) Evaluate the licensee's performance by circling the appropriate statements below and justifying that statement in the appropriate block to the right of that statement. If it is not applicable, put "NA" in the JUSTIFICATION BLOCK.

(3) Complete this evaluation at the end of each inspection and attach to the inspection report. The branch secretary will file these by licensee. Branch personnel should refer to these evaluations when assembling SALP input

1. MANAGEMENT INVOLVEMENT AND CONTROL IN ASSURING QUALITY

Category 1	Category 2	Category 3	JUSTIFICATION (Were there violations or deviations? Did licensee commit to make a change? Did any specific incidents occur to justify your evaluation? Has this occurred before?)
consistent evidence of prior planning and assignment of priorities; well stated, controlled and explicit procedures for control of activities	evidence of prior planning and assignment of priorities; stated, defined procedures for control of activities	little evidence of prior planning and assignment of priorities; poorly stated or ill understood procedures for control of activities	[redacted]
well stated, disseminated and understandable policies	adequately stated and understood policies	poorly stated, poorly understood or non-existent policies	
decision making consistently at a level that ensures adequate management review	decision making usually at a level that ensures adequate management review	decision making seldom at a level that ensures adequate management review	
corporate management frequently involved in site activities	corporate management usually involved in site activities	corporate management seldom involved in site activities	
audits complete, timely and thorough	audits generally complete, and thorough	audits frequently not timely, incomplete, or not thorough	
committees properly staffed and functioning in almost all cases	committees usually properly staffed and functioning	committees not properly staffed or functioning	
reviews timely, thorough and technically sound	reviews generally timely, thorough and technically sound	reviews not timely, thorough or technically sound	
records complete, well maintained and available	records generally complete, well maintained and available	records not complete, not well maintained or unavailable	
procedures and policies strictly adhered to	procedures and policies rarely violated	procedures and policies occasionally violated	
corrective action systems promptly and consistently recognize and address non-reportable concerns	corrective action systems generally recognize and address non-reportable concerns	corrective action systems rarely recognize and address non-reportable concerns	
procurement well controlled and documented	procurement generally well controlled and documented	repetitive breakdown in procurement control	
design well controlled and verified	rare breakdowns of minor significance in design control or verification	repetitive breakdown in designs control or verification	

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June 82

POST INSPECTION SALP EVALUATION

2. APPROACH TO RESOLUTION OF TECHNICAL ISSUES FROM A SAFETY STANDPOINT

Category 1	Category 2	Category 3	JUSTIFICATION
clear understanding of issues demonstrated	understanding of issues generally apparent	understanding of issues frequently lacking	[redacted]
conservatism routinely exhibited when potential for safety significance exists	conservatism generally exhibited	meets minimum requirements	
technically sound and thorough approaches in almost all cases	viable and generally sound and thorough approaches	often viable approaches, but lacking in thoroughness or depth	
timely resolutions in almost all cases	generally timely resolutions	resolutions often delayed	

3. RESPONSIVENESS TO NRC INITIATIVES

Category 1	Category 2	Category 3	JUSTIFICATION
meets deadlines	generally timely responses	frequently requires extensions of time	[redacted]
timely resolution of issues	few longstanding regulatory issues attributable to licensee	longstanding regulatory issues attributable to licensee	
technically sound and thorough responses in almost all cases	viable and generally sound and thorough responses	often viable responses, but lacking in thoroughness or depth	
acceptable resolutions proposed initially in most cases	acceptable resolutions generally proposed	considerable NRC effort or repeated submittals needed to obtain acceptable resolutions	

4. ENFORCEMENT HISTORY

Category 1	Category 2	Category 3	JUSTIFICATION
major violations are rare and are not indicative of programmatic breakdown	major violations are rare and may indicate minor programmatic breakdown	multiple major violations or programmatic breakdown indicated	[redacted]
minor violations are not repetitive and not indicative of programmatic breakdown	multiple minor violations or minor programmatic breakdown indicated	minor violations are repetitive and indicative of programmatic breakdown	
corrective action is prompt and effective	corrective action is timely and effective in most cases	corrective action is delayed or not effective	

POST INSPECTION SALP EVALUATION

5. REPORTING AND ANALYSIS OF REPORTABLE EVENTS

Category 1	Category 2	Category 3	JUSTIFICATION
events promptly and completely reported	events are reported in a timely manner, some information may be lacking	event reporting is frequently late or incomplete	[redacted]
events are properly identified and analyzed	events are accurately identified, some analyses are marginal	events are poorly identified or analyses are marginal, events are associated with programmatic weaknesses	
corrective action is effective as indicated by lack of repetition	corrective action is usually taken but may not be effective as indicated by occasional repetition	corrective action is not timely nor effective, events are repetitive	

6. STAFFING (INCLUDING MANAGEMENT)

Category 1	Category 2	Category 3	JUSTIFICATION
positions are identified, authorities and responsibilities are well defined	key positions are identified, and authorities and responsibilities are defined	positions are poorly identified, or authorities and responsibilities are ill-defined	[redacted]
vacant key positions are filled on priority basis	key positions usually filled in a reasonable time	key positions are left vacant for extended periods of time	
staffing is ample as indicated by control over backlog and overtime	staffing is adequate, occasional difficulties with backlog or overtime	staffing is weak or minimal as indicated by excessive backlog and overtime	

7. TRAINING AND QUALIFICATION EFFECTIVENESS

Category 1	Category 2	Category 3	JUSTIFICATION
training and qualification program makes a positive contribution, commensurate with procedures and staffing, to understanding of work and adherence to procedures with few personnel errors	training and qualification program contributes to an adequate understanding of work and fair adherence to procedures with a modest number of personnel errors	training and qualification program is found to be the major contributing factor to poor understanding of work, as indicated by numerous procedure violations or personnel errors	[redacted]
training program is well defined and implemented with dedicated resources and a means for feed back experience; program is applied to nearly all staff	a defined program is implemented for a large portion of the staff	program may be either lacking, poorly defined, or ineffectively applied for a significant segment of the staff	

POST INSPECTION SALP EVALUATION

FACILITY NMP #2 LICENSEE NMPC INSP. REPORT NO. 85-48 INSPECTION DATE(s) 12-16-19, 85

INSPECTOR Bailey / Norman REVIEWED BY R.R. Keimig DATE

6/2

INSTRUCTIONS:

(1) Indicate the functional area inspected ______________ (operations, refueling, maintenance, etc.). If the inspection covered more than one functional area, complete another Post Inspection SALP Evaluation for that area.

(2) Evaluate the licensee's performance by circling the appropriate statements below and justifying that statement in the appropriate block to the right of that statement. If it is not applicable, put "NA" in the JUSTIFICATION BLOCK.

(3) Complete this evaluation at the end of each inspection and attach to the inspection report. The branch secretary will file these by licensee. Branch personnel should refer to these evaluations when assembling SALP input

1. MANAGEMENT INVOLVEMENT AND CONTROL IN ASSURING QUALITY

Category 1	Category 2	Category 3
consistent evidence of prior planning and assignment of priorities; well stated, controlled and explicit procedures for control of activities	evidence of prior planning and assignment of priorities; stated, defined procedures for control of activities	little evidence of prior planning and assignment of priorities; poorly stated or ill understood procedures for control of activities
well stated, disseminated and understandable policies	adequately stated and understood policies	poorly stated, poorly understood or non-existent policies
decision making consistently at a level that ensures adequate management review	decision making usually at a level that ensures adequate management review	decision making seldom at a level that ensures adequate management review
corporate management frequently involved in site activities	corporate management usually involved in site activities	corporate management seldom involved in site activities
audits complete, timely and thorough	audits generally complete, and thorough	audits frequently not timely, incomplete or not thorough
committees properly staffed and functioning in almost all cases	committees usually properly staffed and functioning	committees not properly staffed or functioning
reviews timely, thorough and technically sound	reviews generally timely, thorough and technically sound	reviews not timely, thorough or technically sound
records complete, well maintained and available	records generally complete, well maintained and available	records not complete, not well maintained or unavailable
procedures and policies strictly adhered to	procedures and policies rarely violated	procedures and policies occasionally violated
corrective action systems promptly and consistently recognize and address non-reportable concerns	corrective action systems generally recognize and address non-reportable concerns	corrective action systems rarely recognize and address non-reportable concerns
procurement well controlled and documented	procurement generally well controlled and documented	repetitive breakdown in procurement control
design well controlled and verified	rare breakdowns of minor significance in design control or verification	repetitive breakdown in designs control or verification

JUSTIFICATION

(Were there violations or deviations? Did licensee commit to make a change? Did any specific incidents occur to justify your evaluation? Has this occurred before?)

[redacted]

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 5
FOIA- 91-A-2 (90-249) B-5

Region 1 Form 197
June 82

POST INSPECTION SALP EVALUATION

2. APPROACH TO RESOLUTION OF TECHNICAL ISSUES FROM A SAFETY STANDPOINT

Category 1	Category 2	Category 3	JUSTIFICATION
clear understanding of issues demonstrated	understanding of issues generally apparent	understanding of issues frequently lacking	[redacted]
conservatism routinely exhibited when potential for safety significance exists	conservatism generally exhibited	meets minimum requirements	
technically sound and thorough approaches in almost all cases	viable and generally sound and thorough approaches	often viable approaches, but lacking in thoroughness or depth	
timely resolutions in almost all cases	generally timely resolutions	resolutions often delayed	

3. RESPONSIVENESS TO NRC INITIATIVES

Category 1	Category 2	Category 3	JUSTIFICATION
~~meets deadlines~~	generally timely responses	frequently requires extensions of time	[redacted]
~~timely resolution of issues~~	few longstanding regulatory issues attributable to licensee	longstanding regulatory issues attributable to licensee	
technically sound and thorough responses in almost all cases	viable and generally sound and thorough responses	often viable responses, but lacking in thoroughness or depth	
acceptable resolutions proposed initially in most cases	acceptable resolutions generally proposed	considerable NRC effort or repeated submittals needed to obtain acceptable resolutions	

4. ENFORCEMENT HISTORY

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major violations are rare and are not indicative of programmatic breakdown	major violations are rare and may indicate minor programmatic breakdown	multiple major violations or programmatic breakdown indicated	[redacted]
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corrective action is prompt and effective	corrective action is timely and effective in most cases	corrective action is delayed or not effective	

POST INSPECTION SALP EVALUATION

5. REPORTING AND ANALYSIS OF REPORTABLE EVENTS

Category 1	Category 2	Category 3	JUSTIFICATION
events promptly and completely reported	events are reported in a timely manner, some information may be lacking	event reporting is frequently late or incomplete	
events are properly identified and analyzed	events are accurately identified, some analyses are marginal	events are poorly identified or analyses are marginal, events are associated with programmatic weaknesses	
corrective action is effective as indicated by lack of repetition	corrective action is usually taken but may not be effective as indicated by occasional repetition	corrective action is not timely nor effective, events are repetitive	

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positions are identified, authorities and responsibilities are well defined	key positions are identified, and authorities and responsibilities are defined	positions are poorly identified, or authorities and responsibilities are ill-defined	
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staffing is ample as indicated by control over backlog and overtime	staffing is adequate, occasional difficulties with backlog or overtime	staffing is weak or minimal as indicated by excessive backlog and overtime	

7. TRAINING AND QUALIFICATION EFFECTIVENESS

Category 1	Category 2	Category 3	JUSTIFICATION
training and qualification program makes a positive contribution, commensurate with procedures and staffing, to understanding of work and adherence to procedures with few personnel errors	training and qualification program contributes to an adequate understanding of work and fair adherence to procedures with a modest number of personnel errors	training and qualification program is found to be the major contributing factor to poor understanding of work, as indicated by numerous procedure violations or personnel errors	
training program is well defined and implemented with dedicated resources and a means for feed back experience; program is applied to nearly all staff	a defined program is implemented for a large portion of the staff	program may be either lacking, poorly defined, or ineffectively applied for a significant segment of the staff	

PROGRAM OFFICE COMMENTS ON
REGIONAL ENFORCEMENT RECOMMENDATION
EA-89-04

[redacted]

PROGRAM OFFICE: NRR, PDI-1

CONTACT: R. Capra/M. Slosson

DATE: February 23, 1989

Information in this record was deleted
in accordance with the Freedom of Information
Act, exemptions 5
FOIA- 91-A-2 (90-269) B-6 K/7

Docket No. 50-410
License No. NPF-69
EA 89-04

Niagara Mohawk Power Corporation
ATTN: Mr. Lawrence Burkhardt, III
Executive Vice President - Nuclear
301 Plainfield Road
Syracuse, New York 13212

Gentlemen:

Subject: NOTICE OF VIOLATION
(NRC Inspection Report No. 50-410/88-21)

This refers to the special NRC safety inspection conducted on December 3-21, 1988, at Nine Mile Point, Unit 2, Scriba, New York to review the circumstances associated with violations of NRC requirements which were identified by your staff and promptly reported to the NRC, including a violation of a technical specification limiting condition for operation (LCO). The report of the inspection was sent to you on January 9, 1989. On February 2, 1989, an enforcement conference was conducted with you and members of your staff to discuss the violations, their causes, and your corrective actions.

The violations, which are described in the enclosed Notice of Violation, involve: (1) the inoperability of one of the two Automatic Depressurization System (ADS) divisions since initial operation in 1986 (contrary to the technical specifications) because of a wiring error in the logic circuitry which occurred during a construction modification in 1985; and (2) the failure to promptly identify and correct this condition adverse to quality until December 1988, even though opportunities existed to detect this error sooner.

With respect to the second violation, the procedure for performing the preoperational test of the system in May 1986, which should have identified this error, was inadequate and the error went undetected. In addition, in July 1986 during the performance of the ADS logic surveillance test, the specific step of the test procedure which would have identified this wiring error was deferred without proper justification. In May 1988, during performance of this same ADS logic test, a testing anomaly associated with the wiring error was ~~again~~ identified; however, inadequate technical review resulted in failure to obtain proper resolution of the problem. Furthermore, this error was not identified by the Quality Assurance Program during reviews of the "as built" condition of the facility prior to startup, nor during any subsequent reviews of testing.

The NRC recognizes that the safety significance of the inoperable ADS division was minimized because of the operability of other redundant Emergency Core Cooling System (ECCS) equipment, namely, the other automatic ADS trip system, the manual initiation capability of the ADS valves, the High Pressure Core Spray System and the Reactor Core Isolation Cooling System. Nonetheless, [redacted] the two violations are classified in the aggregate as a Severity Level III problem in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1988) (Enforcement Policy). Although a civil penalty is normally considered for a Severity Level III violation or problem, I have considered the escalation and mitigation factors set forth in the enforcement policy and have decided, after consultation with the Director of Enforcement and the Deputy Executive Director for Materials Safety, Safeguards, and Operational Support, not to issue a civil penalty in this case because: (1) the violation was identified by a member of your staff who was persistent and inquisitive in pursuing a problem during a monthly surveillance test that was not specifically designed to detect such errors; and (2) your corrective actions for these violations, when identified, were comprehensive. Although the Division I ADS trip system was inoperable for an extended duration because of failures to identify and correct the error during previous tests, the NRC has decided not to utilize this escalation factor [redacted] since [redacted] the fundamental basis for the NRC decision to classify the two violations in the aggregate as Severity Level III. Furthermore, the enforcement history at Unit 2 is average, as evidenced by Category 2 ratings in the operations and surveillance areas during the last two SALP assessments, and therefore, [redacted]

These changes were instituted to correct

[redacted] The NRC ~~Staff~~ recognizes that Niagara Mohawk Power Corporation (NMPC) has made extensive management and organizational changes within the Nuclear department since Unit 1 went into an extended shutdown in December 1987 for generally poor performance over an extended period. The problems that led to this extended shutdown of Unit 1 included the failures, at multiple levels within the NMPC organization, to promptly identify and correct existing problems. [redacted] developed in response to the Unit 1 shutdown for aggressively improving your organization's ability to resolve existing problems at both units, [redacted] may have contributed to the persistent and inquisitive attitude exhibited by the technician who identified this error in December 1988, [redacted]

You are required to respond to this letter and the enclosed Notice and should follow the instructions specified in the enclosed Notice when preparing your response. In your response, you should document the specific actions taken and any additional actions you plan to prevent recurrence. After reviewing you response to this Notice, including your proposed corrective actions and the results of future inspections, the NRC will determine whether further NRC enforcement action is necessary to ensure compliance with NRC regulatory requirements.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and its enclosure will be place in the NRC Public Document Room.

The responses directed by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

Sincerely,

William T. Russell
Regional Administrator

Enclosure: Notice of Violation

cc w/encl:
C. Mangan, Senior Vice President, Nuclear
J. Willis, General Station Superintendent
C. Terry, Vice President, Nuclear Engineering and Licensing
W. Drews, Maintenance Superintendent
J. A. Perry, Director, Unit 1 Restart Task Force
D. Palmer, Acting Manager, QA
W. Hansen, Manager, Corporate Quality Assurance
R. G. Smith, Unit 2 Superintendent, Operations
R. Randall, Unit 1 Superintendent, Operations
C. Beckham, Manager, Nuclear Quality Assurance Operations
R. B. Abbott, Station Superintendent, Unit 2
K. Dahlberg, Station Superintendent, Unit 1
J. F. Warden, New York Consumer Protection Branch
Connor & Wetterhahn
John W. Keib, Esquire
Department of Public Service, State of New York
State of New York, Department of Law
Director of Power Division
Licensing Project Manager, NRR
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
State of New York

bcc w/encl:
Region I Docket Room (with concurrences)
Management Assistant, DRMA (w/o encl)
DRP Section Chief
Robert J. Bores, DRSS
SECY
J. Taylor, DEDRO
J. Lieberman, OE
W. Russell, RI
D. Holody, RI
L. Chandler, OGC
Enforcement Directors, RII-III
Enforcement Officers, RIV-V
T. Murley, NRR
F. Ingram, PA
J. Bradburne, CA
E. Jordan, AEOD
B. Hayes, OI
S. Connelly, OIA
P. Robinson, OE
S. Varga, NRR
V. Miller, OGP/SP
OE File (3 copies + ltrhd)
EDO Rdg File
DCS
B. Clayton, EDO

Gn5

NOTICE OF VIOLATION

Niagara Mohawk Power Corporation
Nine Mile Point Unit 2

Docket No. 50-410
License No. NPF-69
EA 89-04

During an inspection conducted on December 3-21, 1988, NRC inspectors reviewed the circumstances associated with a violation of a technical specification limiting condition for operation which was identified by the licensee and reported to the NRC. During the inspection, another violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Action," 10 CFR Part 2, Appendix C (Enforcement Policy) (1988), the violations are set forth below:

A. Technical Specification Limiting Condition for Operation (LCO) 3.3.3 requires, in part, that the Emergency Core Cooling System (ECCS) actuation instrumentation shall be operable as shown in Table 3.3.3-1. Table 3.3.3-1 requires, in part, two operable Automatic Depressurization System (ADS) trip channels per trip system whenever the reactor is in Operational ~~Modes~~ Conditions 1, 2 and 3. Technical Specification LCO Action Statement 3.3.3-1.c requires that with either ADS trip system inoperable, the inoperable ADS trip system must be restored to operable status within 7 days providing High Pressure Core Spray System and Reactor Core Isolation Cooling System are operable, or the plant shall be placed in HOT SHUTDOWN within 12 hours.

   Contrary to the above, between November 1986 (initial fuel load) and December 8, 1988, while the reactor was, at times, in Operational ~~Modes~~ Condition 1, 2 or 3, the ADS Division I trip system was inoperable due to a wiring error in the logic circuitry, and the reactor was not placed in the hot shutdown condition.

B. 10 CFR 50, Appendix B, Criterion XVI, requires that measures be established to assure that conditions adverse to quality are promptly identified, and the cause determined and corrected. Niagara Mohawk Power Corporation Quality Assurance Topical Report, Section 16, Corrective Action, also requires that for conditions adverse to quality, appropriate corrective action shall be implemented in a timely manner.

   Contrary to the above, during a modification to the logic circuitry of one of the ADS trip systems in 1985, a wiring error was made that resulted in one of the systems being inoperable (as described in Violation A above), and prior to December 8, 1988, this condition adverse to quality was not identified and corrected, notwithstanding the performance of (1) the preoperational test of the trip system logic circuitry in May 1986, as well as (2) two surveillance tests of the logic circuitry in July 1986 and May 1988.

These violations are categorized in the aggregate as a Severity Level III problem.

Pursuant to the provisions of 10 CFR 2.201, Niagara Mohawk Power Corporation is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, within 30 days of the date of this Notice. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each alleged violation: (1) admission or denial of the alleged violation, (2) the reasons for the violation if admitted, (3) the corrective steps that have been taken and the results achieved, (4) the corrective steps that will be taken to avoid further violations, and (5) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order may be issued to show cause why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Consideration may be given to extending the response time for good cause shown.

FOR THE NUCLEAR REGULATORY COMMISSION

William T. Russell
Regional Administrator

Dated at King of Prussia, Pennsylvania
this day of February 1989

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 2 C 1985

Docket No. 50-410

MEMORANDUM FOR: Richard W. Starostecki, Director
Division of Project and Resident Programs

THRU: Wayne Houston, Deputy Director
Division of BWR Licensing

FROM: Mary F. Haughey, Project Manager
Project Directorate No. 3
Division of BWR Licensing

SUBJECT: NRR SALP INPUT - NINE MILE POINT NUCLEAR STATION UNIT 2

Enclosed is NRR input for the March, 1986 SALP Board meeting for Nine Mile Point Nuclear Station Unit 2. As discussed in the enclosure, our evaluation was conducted according to NRR Office Letter No. 44 dated January 3, 1984 and NRC manual chapter 0516, Systematic Assessment of Licensee Performance.

Mary Haughey
Mary Haughey, Project Manager
Project Directorate No. 3
Division of BWR Licensing

Enclosure:
As stated

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K/3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket No. 50-410

FACILITY: Nine Mile Point Nuclear Station Unit 2

LICENSEE: Niagara Mohawk Power Corporation

EVALUATION PERIOD: February 1, 1985, to January 31, 1986

PROJECT MANAGER: Mary F. Haughey

I. INTRODUCTION

This report contains NRR's input to the SALP review for the Nine Mile Point Nuclear Station Unit 2 (NMP-2). The assessment of the licensee's performance was conducted according to NRR Office Letter No. 44, NRR Inputs to SALP Process, dated January 3, 1984. This Office Letter incorporates NRC Manual Chapter 0516, Systematic Assessment of Licensee Performance.

II. SUMMARY

NRC Manual Chapter 0516 specifies that each functional area evaluated will be assigned a performance category (Category 1, 2, or 3) based on a composite of a number of attributes. The performance of the Niagara Mohawk Power Corporation in the functional area of Licensing Activities is rated Category 2.

III. CRITERIA

The evaluation criteria used in this assessment are given in NRC Manual Chapter 0516 Appendix, Table 1, Evaluation Criteria with Attributes for Assessment of Licensee Performance.

IV. METHODOLOGY

This evaluation represents the integrated inputs of the Licensing Project Manager (LPM) and those technical reviewers who expended significant amounts of effort on NMP-2 licensing actions during the current rating period. Using the guidelines of NRC Manual Chapter 0516, the LPM, each reviewer and their middle management applied specific evaluation criteria to the relevant licensee performance attributes, as delineated in Chapter 0516, and assigned an overall rating category (1, 2, or 3) to each attribute. The reviewers included this information as part of each Safety Evaluation Report transmitted to the Division of Licensing. The LPM, after reviewing the inputs of the technical reviewers, combined this information with her own assessment of licensee performance and, using appropriate weighting factors, arrived at a composite rating for the applicant. A written evaluation was then prepared by the LPM and circulated to NRR management for comments. These comments were incorporated in the final draft.

The basis for this appraisal was the applicant's performance in support of licensing actions that were either completed or had a significant level of activity during the current rating period. These actions are as follows:

(1) Responses to the staff requests for information.

(2) Responses to outstanding and confirmatory issues in the SER.

(3) Presentations, responses and support for the ACRS full and subcommittee meetings.

(4) Support for NRR on-site audits during the SALP period.

(5) Response to the downcomer supports issue.

(6) Support of the Technical Specification review.

V. ASSESSMENT OF PERFORMANCE ATTRIBUTES

The applicant's performance evaluation is based on a consideration of five of the seven attributes specified in NRC Manual Chapter 0516. These are:

-- Management Involvement and Control in Assuring Quality

-- Approach to Resolution of Technical Issues from a Safety Standpoint

-- Responsiveness to NRC Initiatives

-- Staffing

-- Training

For the remaining two attributes (enforcement and reportable events), no basis exists for an NRR evaluation for the functional area of Licensing Activities.

Licensing Activities

1. Management Involvement and Control in Assuring Quality

2. Approach to Resolution of Technical Issues from a Safety Standpoint

3. Responsiveness to NRC Initiatives

4. Enforcement History

No basis exists for an NRR evaluation for the functional area of Licensing Activities.

5. Reporting and Analysis of Reportable Events

No basis exists for an NRR evaluation for the functional area of Licensing Activities.

6. Staffing

7. Training

8. Conclusion

Other Review Areas (follows J. Linville memo 1/9/86)

1. Operations

2. Training

See the same subject in the licensing area.

3. Radiological Controls

4. Maintenance

5. PSI/ISI Program and Performance

6. Preoperational Testing

7. Fire Protection

8. Security

9. Construction

10. Quality Programs and Contro's

No NRR input for this area.

Information to be Added to
Section V of the SALP Report - "Supporting Data and Summary"

1. NRR Licensee Meetings

A large number of meetings were held with the applicant in Bethesda to resolve/discuss staff concerns. These are documented by meeting summaries.

2. NRR Site Visits & Audits

Instrumentation and Control Audit	January 7, 8, & 9, 1986
Environmental Qualification Audit	December 16 - 20, 1985
Seismic Qualification Review Team Audit	July 8 - 12, 1985
Pump and Valve Operability Review Team Audit	July 8 - 12, 1985
Containment Systems Site Visit	January 7, 1986
Electrical Power Systems Site Visit	December 17 & 18, 1985
DCRDR Audit	March 19 - 22, 1985
SPDS Audit	July 17 & 18, 1985
Revetment Ditch Audit	August 27, 1985

3. Licensing Documents Issued

FES	April 1985
SER	February 1985
SSER-1	June 1985
SSER-2	November 1985
Draft Technical Specifications	August 29, 1985
Proof-and-Review Technical Specifications	November 20, 1985

4. Applicant Responses

   a. Responses to requests for information.

   b. Letters & FSAR updates to respond to SER concerns.

   c. Responses to ACRS questions.

   d. Responses to concerns on downcomer supports.

5. Support for the Technical Specification review.

6. Support for the ACRS full and subcommittee meetings.

HISTORY OF SALP RATINGS FOR
THE PREVIOUS TWO RATING PERIODS

October 1982 - September 1983

SUMMARY OF RESULTS

NINE MILE POINT, UNIT 2

Functional Areas	Category 1	Category 2	Category 3
Soils and Foundations	X		
Containment and Other Safety Related Structures		X	
Piping Systems and Supports			X
Safety Related Components		X	
Support Systems		No basis for rating	
Electrical Power Supply and Distribution		X	
Instrumentation and Control Systems		X	
Licensing Activities		X	
Project Management/Quality Assurance			X

October 1983 - January 1985

Functional Area	Category Last Period	Category This Period	Recent Trend
	(10-1-82 - 9-30-83)	(10-1-83 - 1-31-85)	
Containment and other Safety Related Structures	2	2	Consistent
Piping Systems and Supports	3	2	Improving
Safety Related Components-Mechanical	2	1	Consistent
Support Systems	Not Assessed	1	Consistent
Electrical Equipment and Cables	2	3	Consistent
Instrumentation and Control Systems	2	2	Consistent
Licensing Activities	2	2	Consistent
Project Management/Quality Assurance	3	2	Improving
Nondestructive Examination	Not Assessed	2	Improving
Engineering	Not Assessed	3	Improving

NMP-2 SALP (Feb. 1985 - Jan. 1986)

MATRIX OF REVIEW BRANCH INPUTS

CRITERIA

Reviewer	Branch	Date	1	2	3	4	5	6	7
R. Wright	EQB								
D. Smith	MTEB								
F. Witt	CHEB								
R. Benedict	LQB								
B. Elliot	MTEB								
*F. Witt	CHEB								
J. Lane	CSB								
K. Desai	RSB								
A. Singh	ASB								
J. Read	AEB								
*J. Lane	CSB								
J. Mauck	ICSB								
Lomb./Romney	EQB								
M. Hum	MTEB								
S. Saba	HFEB								
R. Manili	NMSS								
J. Kudrick	CSB								
Average									

I = insufficient input
NA = not applicable
* = input from same reviewer not counted twice

CRITERIA

The following criteria were used as applicable in evaluation of each functional area:

1. Management involvement in assuring quality.

2. Approach to resolution of technical issues from a safety standpoint.

3. Responsiveness to NRC initiatives.

4. Enforcement history

5. Reporting and analysis of reportable events.

6. Staffing (including management).

7. Training effectiveness and qualification.

To provide consistent evaluation of licensee performance, attributes associated with each criterion and describing the characteristics applicable to Category 1 and 2 and 3 performance were applied as discussed in NRC Manual Chapter 0516, Part II and Table 1.

NEWMAN & HOLTZINGER, P.C.

1615 L STREET, N.W.

WASHINGTON, D.C. 20036

202-955-6600

January 14, 1991

APPEAL OF INITIAL FOIA DECISION
91A2E&C (FOIA-90-269)
91A3E (FOIA-90-270
Rec'd 1-17-91

CERTIFIED MAIL--RETURN RECEIPT REQUESTED

Executive Director for
Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Appeal From Initial FOIA Decision:
FOIA-90-269; FOIA-90-270

Dear Sir or Madam:

Pursuant to 10 C.F.R. § 9.27, please be advised that we hereby appeal several decisions of the NRC FOIA office with regard to the release of documents covered by FOIA-90-269 and FOIA-90-270. In total, this production includes Releases A through K for FOIA-90-269 (spanning the dates August 22, 1990 through December 13, 1990), and Releases A through L for FOIA-90-270 (spanning August 3, 1990 through October 31, 1990). Pursuant to my understanding with Mr. John Phillips, the time for appealing denials contained in any of the releases for either request runs from the date of the final release for both requests (please see my letter to Mr. Phillips dated October 24, 1990).

A. Items Described In The FOIA Requests Which Should Have Yielded Responsive Agency Records

There are a number of inspections, meetings, and other items described in FOIA-90-269 and FOIA-90-270 for which no

~~9102250292~~

responsive documents were produced, yet for which it would appear that such documents should exist.

1. FOIA-90-269

Reconciliation of what we received under this request with what we asked for indicates that we did not receive the following inspection reports:

Inspection Report No.	Date
81.04	04/22/81
82.06	06/01/82
85.15	06/11-06/19/85
85.41	12/10-12/19/85

Please double-check your files and provide us copies of the reports identified above along with copies of all documents relating thereto, as set forth in more detail in our initial request. If copies of these inspection reports do not exist, please confirm that fact.

2. FOIA-90-270

Similarly, a reconciliation of the documents released to us under this request with the documents we requested indicates that there are a number of items for which no responsive documents were produced. In particular, we did not receive any documents responsive to the following items described in Attachment A to FOIA-90-270:

Para. No.	Para. No.
1	33
2	34
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Please review your files again (including those of Region I) and determine whether there exist agency records responsive to the foregoing categories of documents. If for a particular category no responsive documents exist, please state that as well. As previously discussed with the FOIA office, to the extent that a final SALP report is responsive to a request, such reports need not be produced as we already have them.

B. Overly Restrictive Definition Of "Agency Records"

In both FOIA-90-269 and FOIA-90-270, we requested a number of documents contained in the personal files of inspectors, meeting attendees, and others to the extent that such documents were responsive to our specific requests. In its July 12, 1990 response and fee estimate, the FOIA office noted that only "agency records" are releasable, and noted further that inspectors' notes, to the extent that such notes were neither circulated nor required by the agency, are not agency records. Such a position is correct only as far as it goes. Certainly, when we requested documents from an individual's personal files, we were not seeking any documents related to personnel matters, nor were we looking for documents of a private or personal nature. We sought only agency records which were put to agency use.

We believe, however, that the FOIA office used an overly restrictive approach in determining which documents constituted agency records. We request that you reexamine the guidelines used by the FOIA office in this regard, keeping in mind the relevant authorities concerning the definition of the term "agency record." For example, in Bureau of National Affairs v. U.S. Department of Justice, 742 F.2d 1484, 1493 (D.C. Cir. 1984), the court noted that an inquiry was required:

> into the purpose for which the document was created, the actual use of the document, and the extent to which the creator of the document and other employees acting within the scope of their employment relied upon the document to carry out the business of the agency.

The Bureau of National Affairs analysis was refined in Washington Post v. U.S. Department of State, 632 F. Supp. 607 (D.D.C. 1986) wherein the court examined the totality of the circumstances underlying the creation of the document by reference to a four-factor test considering whether: (a) the documents were generated within the agency; (b) the documents have been placed into agency files; (c) the documents are within the agency's control; and (d) the documents have been used by the agency for agency purpose. 632 F. Supp. at 612.

Please ensure that all agency records, wherever they were maintained (including inspectors' personal files), were disclosed, and that no such documents were improperly withheld.

C. Specific Appeals of Agency Records That Were Withheld In Their Entirety Or Released In Part

In the multiple releases to both FOIA-90-269 and FOIA-90-270, the FOIA office frequently invoked the various FOIA exemptions as grounds for withholding an agency record either in part or in full. In many cases, however, a review of the description of the document (in the case of a document withheld in its entirety) or the review of text surrounding a redaction (in the case of a document released in part) strongly suggests that the FOIA office was overly restrictive in its use of these exemptions, and thereby improperly shielded segregable fact from disclosure. Accordingly, for each of the documents listed below, we request that you reexamine the decision of the FOIA office, and release those documents, or portions thereof, which were improperly withheld.

The exemption most commonly invoked by the FOIA office in response to FOIA-90-269 and FOIA-90-270 is the deliberative process privilege embodied in Exemption 5. 1/ That privilege exempts the disclosure of predecisional information which would tend to inhibit the open and frank exchange of ideas essential to the deliberative process. However, even if an agency record is predecisional, "the privilege applies only to the 'opinion' or 'recommendatory' portion of [a document], not to factual information which is contained in the document." Coastal States Gas Corp. v. Department of Energy, 617 F.2d 854, 867 (D.C. Cir. 1980).

Additionally, courts do not look favorably on efforts by an agency to shield from disclosure factual information based on the agency's erroneous belief that such factual disclosure

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1/ It is only this exemption with which this appeal is concerned. Although Exemptions 3, 4, 6, and 7 were used by the FOIA office as grounds for withholding some or all of various other documents, we are not appealing the invocation of such exemptions for those other documents.

would also disclose the deliberative process of the agency. For example, in City of Chicago v. U.S. Nuclear Regulatory Commission, 547 F. Supp. 740 (N.D. Ill. 1982), the city brought suit to compel the disclosure of a draft environmental impact statement and related documents. With regard to the NRC's argument that release of factual material contained in predecisional records would reveal the essence of the deliberative process, the court stated:

> We think the broad, implicit suggestion of the NRC in this case that revelation of the facts contained in early versions of the Draft EIS would reveal the 'essence of the deliberative process' may prove too much. Unless the NRC can provide some context for its conclusory suggestion that the deliberative process will be revealed through disclosure of factual portions of the preliminary documents, the effect of their argument would be to make all factual portions of preliminary documents immune from disclosure in contravention of the contemplation of the Supreme Court in EPA v. Mink and the terms of the statute, 5 U.S.C. § 552(a)(4)(B), providing for partial disclosure of documents.

547 F. Supp. at 749 (footnote omitted). See also Moore-McCormick Lines, Inc. v. ITO Corp. of Baltimore, 508 F.2d 945 (4th Cir. 1974) (inferences based on observed facts which depend on the expertise of the investigating official not subject to withholding as exempt).

In reviewing the merits of the following appeals, please keep in mind that "[t]he established doctrine is that the [FOIA] Exemptions are to be construed narrowly." City of Chicago v. U.S. Nuclear Regulatory Commission, 547 F. Supp. at 740. The specific decisions we wish to appeal are as follows:

1. FOIA-90-269

(a) Appendix D; item 1; released in part under Exemption 5 (deliberative process)

This document consists of a partially redacted cover memo from Thomas E. Murley to J. Taylor dated January 20, 1987 concerning a proposed enforcement action for Nine Mile Point 2 (NMP-2), accompanied by a draft letter and Notice of Violation which was withheld in their entirety (*see* Attachment A hereto.)

It is clear from the text of the document surrounding the redactions that the redacted sections likely contain factual matters which should have been disclosed. For example, on the first page of the memo, the two paragraphs not redacted concern violations involving a Source Range Monitor and the bypassing of a Source Range Monitor downscale rod. There is no reason to believe that the redacted portions of the document would reflect anything other than factual concerns of the same nature as those expressed in the portions of the document that were not redacted. In addition, the draft letter and Notice of Violation, which were attached to the memo but which were withheld from disclosure in their entirety, also should be disclosed for the same reasons.

(b) Appendix G; items 1, 2, and 6; documents withheld in their entirety under Exemptions 3, 4, and 5 (deliberative process).

Here, we are appealing the decision to withhold in their entirety items 1, 2, and 6 of Appendix G. Items 1 and 2 are post-inspection worksheets, and item 6 is a memo from W.T. Russell to James Lieberman dated April 7, 1988 concerning a proposed letter to Niagra Mohawk Power Corp. (NMPC) regarding the findings of an OI investigation. Please note that we are not appealing the withholding of items 3, 4, and 5, concerning draft input to various SALP reports on reactor safeguards. Presumably, the FOIA office's reliance on Exemption 3 (unclassified safeguards information) and Exemption 4 (proprietary information pursuant to 10 C.F.R. 2.790(d)(1)) concerns items 3, 4, and 5 only.

With regard to the inspection worksheets identified in items 1 and 2, it appears highly likely that the worksheets contain disclosable fact, not opinion. To the extent that these worksheets contain entries reflecting the agency's deliberative process, such entries could be redacted. Similarly, it appears doubtful that the memo identified in item 6 embraces matters relating solely to the deliberative process of the agency.

Certainly, that memo sets forth facts and findings based on fact, and, at a minimum, should have been disclosed to that extent.

(c) Appendix K; items 1, 3, 5, and 7; documents released in part under Exemptions 3, 4, and 5 (deliberative process).

For this release, we are appealing the decision to withhold portions of items 1, 3, 5, and 7 under claim of the deliberative process privilege embodied in Exemption 5 (we are not appealing the redactions contained in items 2 and 4, made pursuant to Exemptions 3 and 4). For each of the redactions contained in items 1, 3, 5, and 7, we believe that the FOIA office improperly withheld factual matters under the guise of the deliberative process exemption.

The first item of Appendix K is a memo from Mary Haughey to Darrell Eisenhut dated October 28, 1983 concerning the NRR input to the NMP-2 SALP report (see Attachment B). When the redactions are placed in the context of the rest of the memorandum, it appears clear that the sections redacted were statements of fact, not opinion. For example, entire sections are redacted under the headings in the body of the document which read "Management Involvement and Control in Assuring Quality," "Responsiveness to NRR Initiatives," "Staffing," and other such headings which would clearly suggest that the text accompanying these headings sets forth factual findings, not opinions pursuant to the deliberative process. Accordingly, this document should be reissued in its unredacted state.

Item 3 of Appendix K, which is a memo from Mary Haughey to Richard Starostecki dated February 23, 1986 concerning the NRR input to a SALP report for NMP-2 was also severely redacted under claim of Exemption 5 (see Attachment C). Again, the FOIA office redacts entire textual sections following headings such as "Management Involvement and Control in Assuring Quality," "Approach to Resolution of Technical Issues from a Safety Standpoint," "Responsiveness to NRC Initiatives," "Staffing," "Training," and like headings. Again, when placed in the context of the document as a whole, the redactions appear to have improperly withheld facts or findings based on facts, not opinions pursuant to the deliberative process.

Also released in part pursuant to Exemption 5 is a memo dated May 20, 1988 concerning a proposed letter to NMPC concerning the submittal of incomplete and misleading information to the NRC (see Attachment D). For this document, entire pages have been redacted, which, in itself, suggests that the FOIA office was overly zealous in its invocation of the deliberative process privilege. In addition, when considered in the context of the document as a whole, the redactions would appear to contain facts or findings relating to the cable pull problem and the extent to which this problem violated the Quality Control procedures. At a minimum, the document should have been released in a less severely redacted state.

Item 7 of Appendix K, which is a memo from Steven Vargo to James Lieberman dated February 28, 1989 concerning a proposed civil penalty for NMP-2, was also partially withheld pursuant to Exemption 5 (see Attachment E). What was released of this document was the cover sheet only, not the report that was presumably transmitted therewith. While it is difficult to evaluate the basis for withholding a report without the benefit of seeing the report itself, the FOIA office presumably was persuaded by the legend appearing on the transmittal sheet that reads "Sensitive-Predecisional Information-Internal Distribution Only." Surely, the report, which was redacted in its entirety, contains factual matters subject to disclosure.

2. FOIA-90-270

(a) Appendix G; item 2; released in part under Exemption 5 (deliberative process

Identified as item 1 to this appendix is a memo from Daniel Holody to Jane Axelrad dated February 25, 1983 concerning a proposed civil penalty for NMP-2. When considered in the context of the document as a whole, the redactions in the memo appear to withhold findings based on fact, not opinion pursuant to the deliberative process. Since the items specifically enumerated in the cover sheet concern site deficiencies at NMP-2, there is no reason to believe that the redacted paragraphs address anything other than additional site deficiencies. Similarly, the first attachment to that memo, the proposed letter and Notice of Violation (which was withheld in its entirety), also likely contains segregable fact subject to disclosure.

(b) Appendix H; items 1 and 2; withheld in their entirety under Exemption 5 (deliberative process)

Item 1 is described as a memo dated August 4, 1983 from Daniel Holody to Jane Axelrad concerning the proposed order imposing a civil penalty for NMP-2. Although there is no sure way of knowing the contents of the document, as the document was withheld in its entirety, it would appear that document contains statements of fact and agency findings based on fact. Consequently, it would appear that a blanket withholding of the document based on a claim of privilege under Exemption 5 is improper. Similarly, it appears that the withholding of the document described in Item 2 of Appendix H is also improper. That document is a memorandum from Daniel Holody to Jane Axelrad dated December 14, 1983 concerning a proposed Notice of Violation for NMP-2. For the same reasons as those set forth for Item 1, the blanket withholding of this document also appears to be improper.

* * *

For the foregoing reasons, we request that you give these matters your thorough and prompt consideration and that you direct the FOIA office to release all additional documents that are responsive to FOIA-90-269 and FOIA-90-270. Please provide your response along with copies of any such additional documents within the 20-day period set forth in the regulations. Please let me know if you have any questions concerning this matter.

Sincerely,

John C. Person

cc: Mr. John Phillips
Judith A. Lockhart, Esq.
Thomas Carruthers, Esq.
Kevin P. Gallen, Esq.

NOT PUBLICLY RELEASED

POLICY ISSUE

(NEGATIVE CONSENT)

May 20, 1988

SECY-88-137

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: PROPOSED LETTER TO NIAGARA MOHAWK POWER CORPORATION (NMPC) BASED ON THE SUBMITTAL OF INCOMPLETE AND MISLEADING INFORMATION TO THE NRC CONCERNING AN INTERNAL INVESTIGATION PERFORMED BY THE COMPANY AT NINE MILE POINT, UNIT 2

Purpose: To advise the Commission of the staff's intention to issue the enclosed letter to Niagara Mohawk Power Corporation based on the licensee's submittal to the NRC of a letter containing incomplete and misleading information concerning an internal investigation performed by the company at Nine Mile Point, Unit 2.

Discussion: In a memorandum dated May 8, 1987, the Director, Office of Investigations (OI), provided me and the Regional Administrator, RI, a copy of the subject OI investigation report, dated March 12, 1987, entitled "Nine Mile Point, Unit 2 (NMP-2): Alleged Material False Statements Regarding Results of Licensee Quality First Investigation." The OI investigation was conducted in response to a request, dated October 3, 1985, from the then Regional Administrator (RA), RI to the Director, OI:RI, that OI determine whether the licensee's Quality First Program (Q1P) investigation of alleged inadequate nuclear instrument cable pulls at Unit 2 in May 1985, accurately reported the facts of this incident in a letter to Region I, dated July 11, 1985. The basis for the RA request was an allegation received in September 1985, via the State of New York Public Service Commission, that the Q1P investigation, conducted in June-July 1985, was inadequate, and events associated with the cable installation were different than described in the Q1P investigation.

The licensee's Q1P investigation was conducted in response to a Region I request to the licensee in a letter dated June 7, 1985. The request was made after Region I had received an anonymous allegation on May 28, 1985 concerning the adequacy of the installation of the neutron monitoring system (NMS) cables. Specifically, the alleger indicated that two of the cables were broken during

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 5
FOIA 90-249

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Contact: James Lieberman, OE
20741
Geoffrey Cant, OE
23283

K/5

installation due to excessive pull tension on the cables; contractor Quality Control (QC) inspectors observed this activity without issuing any unsatisfactory inspection report; and supervisors improperly pressured individuals to complete the installation effort. In the June 7 letter, Region I requested that the licensee's investigation determine the technical adequacy of the installed cable, the adequacy of the QC coverage of the activity, and the measures taken to ensure that no improper pressure to complete work occurred.

Upon completion of its QIP investigation, the licensee responded to the NRC with its July 11, 1985 letter, signed by the Director of Quality Assurance (QA), which stated that the "installation of the cable pull was technically satisfactory, the Quality Control program was adhered to and no evidence of improper pressure apparently by supervision was found." Attached to that letter was a two page report of the QIP investigation which formed the basis for the licensee's conclusions.

As a result of these conclusions, OI referred its investigation report to the Department of Justice on May 8, 1987, and indicated that the information contained in the July 11 letter, as well as the regulatory violations committed during the NMS cable installation effort, were indicative of possible violations of criminal law. In a memo dated November 6, 1987, DOJ informed the NRC that the Department declined to prosecute this matter.

As a result of these procedural violations, Region I sent the licensee a Notice of Violation on January 22, 1986 to address these concerns and solicit appropriate corrective actions. The corrective actions for these procedural violations were provided in a letter dated March 3, 1986.

Coordination: The Office of the General Counsel has no legal objection.

Recommendation:

Victor Stello, Jr.
Executive Director for Operations

Enclosure:
As Stated

SECY NOTE: In the absence of instructions to the contrary, SECY will notify the staff on Tuesday, June 7, 1988 that the Commission, by negative consent, assents to the action proposed in this paper.

DISTRIBUTION:
Commissioners
OGC
OI
OIA
GPA
REGION I
EDO
SECY

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST

RESPONSE TYPE

FINAL | X PARTIAL

DATE SEP 2 0 1990

DOCKET NUMBER(S) (If applicable)

REQUESTER

John C. Person

PART I. - AGENCY RECORDS RELEASED OR NOT LOCATED (See checked boxes)

	No agency records subject to the request have been located.
	No additional agency records subject to the request have been located.
	Requested records are available through another public distribution program. See Comments Section.
X	Agency records subject to the request that are identified on Appendix(es) B are already available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC 20555.
X	Agency records subject to the request that are identified on Appendix(es) C are being made available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
	The nonproprietary version of the proposal(s) that you agreed to accept in a telephone conversation with a member of my staff is now being made available for public inspection and copying at the NRC Public Document Room 2120 L Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
	Agency records subject to the request that are identified on Appendix(es) ______ may be inspected and copied at the NRC Local Public Document Room identified in the Comments Section.
	Enclosed is information on how you may obtain access to and the charges for copying records placed in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC.
	Agency records subject to the request are enclosed.
	Records subject to the request have been referred to another Federal agency(ies) for review and direct response to you.
	You will be billed by the NRC for fees totaling $ ______
	In view of NRC's response to this request, no further action is being taken on appeal letter dated ______ No. ______

PART II. A—INFORMATION WITHHELD FROM PUBLIC DISCLOSURE

X	Certain information in the requested records is being withheld from public disclosure pursuant to the exemptions described in and for the reasons stated in Part II, sections B, C, and D. Any released portions of the documents for which only part of the record is being withheld are being made available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC in a folder under this FOIA number and requester name.

COMMENTS

SIGNATURE, DIRECTOR, DIVISION OF FREEDOM OF INFORMATION AND PUBLICATIONS SERVICES

Kathy O. Greene for Donnie H. Grimsley

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NRC FORM 464 (Part I) (1-90)

FREEDOM OF INFORMATION ACT RESPONSE FOIA NUMBER(S) FOIA – [illegible] DATE [illegible]

PART II.B – APPLICABLE EXEMPTIONS

Records subject to the request that are described on the enclosed Appendix(es) D are being withheld in their entirety or in part under the Exemptions and for the reasons set forth below pursuant to 5 U.S.C. 552(b) and 10 CFR 9.17(a) of NRC Regulations.

	1. The withheld information is properly classified pursuant to Executive Order. (EXEMPTION 1)	
	2. The withheld information relates solely to the internal personnel rules and procedures of NRC. (EXEMPTION 2)	
	3. The withheld information is specifically exempted from public disclosure by statute indicated. (EXEMPTION 3)	
		Sections 141-145 of the Atomic Energy Act which prohibits the disclosure of Restricted Data or Formerly Restricted Data (42 U.S.C. 2161-2165).
		Section 147 of the Atomic Energy Act which prohibits the disclosure of Unclassified Safeguards Information (42 U.S.C. 2167).
	4. The withheld information is a trade secret or commercial or financial information that is being withheld for the reason(s) indicated. (EXEMPTION 4)	
		The information is considered to be confidential business (proprietary) information.
		The information is considered to be proprietary information pursuant to 10 CFR 2.790(d)(1).
		The information was submitted and received in confidence pursuant to 10 CFR 2.790(d)(2).
X	5. The withheld information consists of interagency or intraagency records that are not available through discovery during litigation. (EXEMPTION 5). Applicable Privilege:	
	X	Deliberative Process: Disclosure of predecisional information would tend to inhibit the open and frank exchange of ideas essential to the deliberative process. Where records are withheld in their entirety, the facts are inextricably intertwined with the predecisional information. There also are no reasonably segregable factual portions because the release of the facts would permit an indirect inquiry into the predecisional process of the agency.
		Attorney work-product privilege. (Documents prepared by an attorney in contemplation of litigation.)
		Attorney-client privilege. (Confidential communications between an attorney and his/her client.)
	6. The withheld information is exempted from public disclosure because its disclosure would result in a clearly unwarranted invasion of personal privacy. (EXEMPTION 6)	
	7. The withheld information consists of records compiled for law enforcement purposes and is being withheld for the reason(s) indicated. (EXEMPTION 7)	
		Disclosure could reasonably be expected to interfere with an enforcement proceeding because it could reveal the scope, direction, and focus of enforcement efforts, and thus could possibly allow them to take action to shield potential wrongdoing or a violation of NRC requirements from investigators. (EXEMPTION 7(A))
		Disclosure would constitute an unwarranted invasion of personal privacy. (EXEMPTION 7(C))
		The information consists of names of individuals and other information the disclosure of which could reasonably be expected to reveal identities of confidential sources. (EXEMPTION 7(D))
	OTHER	

PART II.C – DENYING OFFICIALS

Pursuant to 10 CFR 9.25(b) and/or 9.25(c) of the U.S. Nuclear Regulatory Commission regulations, it has been determined that the information withheld is exempt from production or disclosure, and that its production or disclosure is contrary to the public interest. The persons responsible for the denial are those officials identified below as denying officials and the Director, Division of Freedom of Information and Publications Services, Office of Administration and Resources Management, for any denials that may be appealed to the Executive Director for Operations (EDO).

DENYING OFFICIAL	TITLE/OFFICE	RECORDS DENIED	APPELLATE OFFICIAL	
			SECRETARY	EDO
Thomas T. Martin	Regional Administrator, Reg. I	D/1		X

PART II.D – APPEAL RIGHTS

The denial by each denying official identified in Part II.C may be appealed to the Appellate Official identified in that section. Any such appeal must be in writing and must be made within 30 days of receipt of this response. Appeals must be addressed as appropriate to the Executive Director for Operations or to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should clearly state on the envelope and in the letter that it is an "Appeal from an Initial FOIA Decision."

Re: FOIA-90-269

APPENDIX B
DOCUMENTS ALREADY AVAILABLE IN THE PDR

NUMBER	DATE	DESCRIPTION
1.	8/21/89	Letter from Steven Varga to Lawrence Burkhardt, subject: NRC Inspection To Review Material Procured From NSJS, Incorporated (50-410/88-201). PDR Accession No. 8908250275

Re: FOIA-90-269

APPENDIX C
DOCUMENTS BEING PLACED IN THE PDR

NUMBER	DATE	DESCRIPTION
1.	4/12/89	Memo from Robert Capra to Bruce Boger, subject: Comparison of the 1986 NMP-1 I&C Technician Allegations With Results of 1989 NMP 1/2 Special Team Inspection Findings. (21 pages)
2.	8/10/89	Letter from E. William Brach to W. L. Strickland, regarding Notice of Nonconformance and Inspection Report 99901117/88-01. (8 pages)

Re: FOIA-90-269

APPENDIX D
DOCUMENTS BEING RELEASED IN PART

NUMBER	DATE	DESCRIPTION
1.	1/20/87	Memo from Thomas E. Murley to J. Taylor, subject: Proposed Enforcement Action - Nine Mile Point, Unit 2, attaching draft letter and Notice of Violation. (7 pages) The attachments, and portions of the cover memo are being withheld pursuant to Exemption 5.

FOIA — 90-269	
RESPONSE TYPE	
FINAL	X PARTIAL
DATE	OCT 1 6 1990
DOCKET NUMBER(S) (If applicable)	

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST

REQUESTER

John Person

PART I.—AGENCY RECORDS RELEASED OR NOT LOCATED *(See checked boxes)*

	No agency records subject to the request have been located.
	No additional agency records subject to the request have been located.
	Requested records are available through another public distribution program. See Comments Section.
X	Agency records subject to the request that are identified on Appendix(es) E are already available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC 20555
X	Agency records subject to the request that are identified on Appendix(es) F are being made available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
	The nonproprietary version of the proposal(s) that you agreed to accept in a telephone conversation with a member of my staff is now being made available for public inspection and copying at the NRC Public Document Room, 2120 L Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
	Agency records subject to the request that are identified on Appendix(es) ____ may be inspected and copied at the NRC Local Public Document Room identified in the Comments Section.
	Enclosed is information on how you may obtain access to and the charges for copying records placed in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC.
	Agency records subject to the request are enclosed.
	Records subject to the request have been referred to another Federal agency(ies) for review and direct response to you.
	You will be billed by the NRC for fees totaling $ ____
	In view of NRC's response to this request, no further action is being taken on appeal letter dated ____ No. ____

PART II. A—INFORMATION WITHHELD FROM PUBLIC DISCLOSURE

X	Certain information in the requested records is being withheld from public disclosure pursuant to the exemptions described in and for the reasons stated in Part II, sections B, C, and D. Any released portions of the documents for which only part of the record is being withheld are being made available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC in a folder under this FOIA number and requester name.

COMMENTS

SIGNATURE, DIRECTOR, DIVISION OF FREEDOM OF INFORMATION AND PUBLICATIONS SERVICES

Donnie H. Grimsley

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FREEDOM OF INFORMATION ACT RESPONSE   FOIA NUMBER(S) FOIA - 89-269   DATE NOV 1 6 1989

PART II.B—APPLICABLE EXEMPTIONS

Records subject to the request that are described on the enclosed Appendix(es) G & H are being withheld in their entirety or in part under the Exemptions and for the reasons set forth below pursuant to 5 U.S.C. 552(b) and 10 CFR 9.17(a) of NRC Regulations.

	1. The withheld information is properly classified pursuant to Executive Order (EXEMPTION 1)	
	2. The withheld information relates solely to the internal personnel rules and procedures of NRC. (EXEMPTION 2)	
X	3. The withheld information is specifically exempted from public disclosure by statute indicated. (EXEMPTION 3)	
		Sections 141-145 of the Atomic Energy Act which prohibits the disclosure of Restricted Data or Formerly Restricted Data (42 U.S.C. 2161-2165)
	X	Section 147 of the Atomic Energy Act which prohibits the disclosure of Unclassified Safeguards Information (42 U.S.C. 2167)
X	4. The withheld information is a trade secret or commercial or financial information that is being withheld for the reason(s) indicated. (EXEMPTION 4)	
		The information is considered to be confidential business (proprietary) information.
	X	The information is considered to be proprietary information pursuant to 10 CFR 2.790(d)(1).
		The information was submitted and received in confidence pursuant to 10 CFR 2.790(d)(2).
X	5. The withheld information consists of interagency or intraagency records that are not available through discovery during litigation. (EXEMPTION 5) Applicable Privilege:	
	X	Deliberative Process: Disclosure of predecisional information would tend to inhibit the open and frank exchange of ideas essential to the deliberative process. Where records are withheld in their entirety, the facts are inextricably intertwined with the predecisional information. There also are no reasonably segregable factual portions because the release of the facts would permit an indirect inquiry into the predecisional process of the agency.
		Attorney work product privilege. (Documents prepared by an attorney in contemplation of litigation.)
		Attorney-client privilege. (Confidential communications between an attorney and his/her client.)
	6. The withheld information is exempted from public disclosure because its disclosure would result in a clearly unwarranted invasion of personal privacy. (EXEMPTION 6)	
	7. The withheld information consists of records compiled for law enforcement purposes and is being withheld for the reason(s) indicated. (EXEMPTION 7)	
		Disclosure could reasonably be expected to interfere with an enforcement proceeding because it could reveal the scope, direction, and focus of enforcement efforts, and thus could possibly allow them to take action to shield potential wrongdoing or a violation of NRC requirements from investigators. (EXEMPTION 7(A))
		Disclosure would constitute an unwarranted invasion of personal privacy. (EXEMPTION 7(C))
		The information consists of names of individuals and other information the disclosure of which could reasonably be expected to reveal identities of confidential sources. (EXEMPTION 7(D))
	OTHER	

PART II.C—DENYING OFFICIALS

Pursuant to 10 CFR 9.25(b) and/or 9.25(c) of the U.S. Nuclear Regulatory Commission regulations, it has been determined that the information withheld is exempt from production or disclosure, and that its production or disclosure is contrary to the public interest. The persons responsible for the denial are those officials identified below as denying officials and the Director, Division of Freedom of Information and Publications Services, Office of Administration and Resources Management, for any denials that may be appealed to the Executive Director for Operations (EDO).

DENYING OFFICIAL	TITLE/OFFICE	RECORDS DENIED	APPELLATE OFFICIAL SECRETARY	APPELLATE OFFICIAL EDO
Thomas T. Martin	Regional Administrator, Reg. I	App. G & H		X

PART II.D—APPEAL RIGHTS

The denial by each denying official identified in Part II.C may be appealed to the Appellate Official identified in that section. Any such appeal must be in writing and must be made within 30 days of receipt of this response. Appeals must be addressed as appropriate to the Executive Director for Operations or to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should clearly state on the envelope and in the letter that it is an "Appeal from an Initial FOIA Decision."

Re: FOIA-90-269

APPENDIX E
DOCUMENTS ALREADY AVAILABLE IN THE PDR

NUMBER	DATE	DESCRIPTION
1.	10/2/86	Letter from C. V. Mangan to E. Adensam, regarding exemption request to permit fuel loading and performance of startup tests. PDR Accession No. 8610100196
2.	10/20/86	Letter from C. V. Mangan to W. Kane, subject: final report on Main Steam Isolation Valve leakage rates. PDR Acc. No. 8610270451
3.	1/13/87	Letter from T. Martin to J. Beratta, regarding site analyses. PDR Acc. No. 8701160101
4.	12/15/87	Letter from T. Martin to J. Haynes, subject: Combined Inspection No. 50-220/86-23; 50-410/86-59. PDR Acc. No. 8712220115.

Re: FOIA-90-269

APPENDIX F
DOCUMENTS BEING PLACED IN THE PDR

NUMBER	DATE	DESCRIPTION
1.	11/29/80	Memo from R. Brady to J. Linville, subject: Trip Report - Nine Mile Pt 2. (2 pages)
2.	7/9/81	Memo from R. A. Feil to H. B. Kister, subject: Status of Inspection Program - Nine Mile Point, Unit 2. (30 pages)
3.	11/9/81	Memo from S. Ebneter to various addressees, subject: Team Inspection, Nine Mile Point 2. (1 page)
4.	6/1/82	Telephone Report, subject: Cable Tray Separation Deficiencies. (1 page)
5.	9/12/83	Memo from T. Martin to D. Eisenhut, subject: Allegations Concerning Adequacy of Standby Diesel Generators at Nine Mile Point, Unit 2. (1 page)
6.	11/3/83	Memo from R. W. Starostecki to T. E. Murley, subject: Revised Nine Mile Point Unit 2 SALP Schedule. (1 page)
7.	11/4/83	Memo from W. Lazarus to D. Haverkamp, subject: Allegation RI-83-A-0019: Failure to Write Nonconformance Reports for Contractor Work During 1979 and 1981 Outages. (1 page)
8.	11/4/83	Memo from J. P. Durr to S. Ebneter, subject: SALP Input (10/1/81 - 9/30/83) for Nine Mile Pt. 2. (3 pages)
9.	2/14/84	Memo from S. Ebneter to various, subject: Construction Appraisal Team Inspection - Nine Mile Pt. 2. (1 page)
10.	2/22/84	List of attendees at Niagara Mohawk/NRC Meeting on 2/22/84. (2 pages)
11.	4/10/84	Memo from A. Varela to S. Collins, subject: Allegation RI-84-A-0004 (Nine Mile 2): Alleger Told to Use Improper Grout Under Main Turbine: After Grout Cracked, Told Not to Inform Others at Nine Mile Pt. 2. (1 page)

Re: FOIA-90-269

APPENDIX F
DOCUMENTS BEING PLACED IN THE PDR
(Continued)

NUMBER	DATE	DESCRIPTION
12.	4/27/84	Memo from R. W. Starostecki to Region I Construction Site Resident Inspectors, subject: Implementation of Augmented Inspection Program - Nine Mile Pt 2. (2 pages)
13.	5/17/84	Pre-Inspection Cover Sheet - Rpt. No. 84-10. (2 pages)
14.	6/21/84	Memo from R. W. Starostecki to T. Murley, subject: Nine Mile Point-Unit 2 Augmented Inspection Program-Proposed. (8 pages)
15.	9/10/84	Memo from R. W. Starostecki to J. N. Grace, subject: Nine Mile Pt. 2 - Management Analysis Company Assessment, Phase I and Phase II Interim Reports, w/o enclosures. (2 pages)
16.	10/10/84	Memo from S. Reynolds to S. Ebneter, subject: SWEC Welder Performance Qualification Testing at Nine Mile Point 2. (8 pages)
17.	2/8/85	Memo from A. Finkel to S. Collins, subject: Division of Reactor Safety Input for Nine Mile Point 2 SALP. (6 Pages)
18.	3/8/85	Memo from R. R. Bellamy to S. J. Collins, subject: Radiological Controls SALP Input For Nine Mile 2. (2 pages)
19.	3/29/85	Region I Enforcement Officer handwritten notes on 1 page of Niagara Mohawk letter. (1 page)
20.	6/11/85	Memo from R. Starostecki to T. Murley, subject: Nine Mile Pt. 2 Augmented Inspection Program Status Update. (1 page)
21.	10/4/85	Memo from R. Starostecki to T. Murley, subject: Nine Mile Pt. Unit [illegible] Augmented Inspection Program Status Update No. 2. (4 pages)

Re: FOIA-90-269

APPENDIX F
DOCUMENTS BEING PLACED IN THE PDR
(Continued)

NUMBER	DATE	DESCRIPTION
22.	12/16/85	Letter from J. Sunser to R. Starostecki, regarding Physical Security Event. (3 pages)
23.	1/27/86	Letter from D. Beckman to J. Linville, subject: NMP-2 Tech Spec Inspection Report - First Draft. (73 pages)
24.	2/10/86	Memo from R. Starostecki to T. Murley, subject: Nine Mile Point Unit 2 Augmented Inspection Program Status Update No. 3. (2 pages)
25.	3/5/86	Memo from R. R. Bellamy to S. J. Collins, subject: Radiological Controls SALP Input - Nine Mi. Pt. 2. (5 pages)
26.	3/7/86	Memo from J. C. Linville, subject: Draft Nine Mile Pt. 2 SALP Board Report. (57 pages)
27.	4/15/86	Memo from J. C. Linville to A. Shropshire, subject: Allegation of Improper RC1 Surveillance Report by Stone and Webster at Nine Mi. Pt. (1 page)
28.	5/7/86	Pre-Inspection Cover Sheet- Rpt. No. 50-410/86-24. (2 pages)
29.	6/26/86	Pre-Inspection Cover Sheet and Inspection Plan - Rpt. 86-35. (1 page)
30.	7/17/86	Pre-Inspection Cover Sheet and Inspection Plan - Rpt. 50-220/86-14, 50-410/86-45. (1 page)
31.	8/6/86	Memo from R. Starostecki to T. Murley, subject: Nine Mile Pt 2 Augmented Inspection Program Status Update No. 4 and Final Rpt. (5 pages)
32.	8/11/86	Pre-Inspection Cover Sheet and Inspection Plan - Rpt. No. 50-410/86-49. (1 page)

APPENDIX F
DOCUMENTS BEING PLACED IN THE PDR
(Continued)

NUMBER	DATE	DESCRIPTION
33.	9/3/86	Memo from L. Bettenhausen to S. Collins, subject: Nine Mi. Pt. 2 Readiness Report Draft. (3 pages)
34.	9/12/86	Memo from R. Gallo to E. Adensam, subject: Closure of Open Licensing Issues at Nine Mi. Pt. 2. (5 pages)
35.	10/7/86	MSIV Highlights. (4 pages)
36.	2/13/87	Memo from P. K. Eapen to R. M. Gallo, subject: Nine Mile Pt. 2, SALP Input for Assessment Period February 1, 1986 Through January 31, 1987. (12 pages)
37.	4/29/87	Memo from J. R. Johnson to R. M. Gallo, subject: Nine Mi. Pt. 2 Initial Criticality Preparation. (1 page)
38.	5/27/87	Memo from J. Johnson to W. F. Kane, subject: Nine Mi. Pt. 2 Operational Readiness Team Inspection. (5 pages)
39.	6/23/87	Memo from W. Kane to S. Varga, subject: Nine Mi. Pt. 2 Licensing Action Related to Deletion of Fire Protection Technical Specifications. (16 pages)
40.	7/2/87	Memo from W. Russell to T. Murley, subject: Nine Mi. Pt. 2 Full Power License Recommendation. (3 pages)
41.	8/19/87	Memo from L. Bettenhausen to R. M. Gallo, subject: Nine Mile-2 Assessment of Performance (Test Condition Heatup). (6 pages)
42.	9/9/87	Memo from L. Wink to R. Gallo, subject: Niagara Mohawk, Nine Mile Pt. 2, Assessment of Licensee Performance (Test Condition 1). (6 pages)
43.	Undated	NMP-2 Enforcement Conference Briefing. (1 page)

Re: FOIA-90-269

APPENDIX F
DOCUMENTS BEING PLACED IN THE PDR
(Continued)

NUMBER	DATE	DESCRIPTION
44.	Undated	10/8/87 Enforcement Conference agenda. (6 pages)
45.	10/19/87	Letter from J. Beratta to T. Martin. (2 pages)
46.	11/9/87	Memo from W. Kane to S. Varga, subject: Licensing Action Review for Nine Mi. Pt. 2 Change to Initial Startup Test Program. (6 pages)
47.	11/12/87	Letter from J. Beratta to W. Kane. (1 page)
48.	1/7/88	Memo from W. Kane to W. Russell, subject: Revised SALP Schedule for Nine Mi. Pt. 1 and 2. (2 pages)
49.	1/13/88	Memo from H. Gray to W. Cook, subject: Feeder report - GL 84-11 Inspection. (3 pages)
50.	1/22/88	Memo from W. Russell to W. Kane, subject: Augmented Inspection Team - Reactor Vessel Overfill Following a Scram at Nine Mi. Pt. 2. (2 pages)
51.	1/26/88	Memo from E. C. Wensinger to W. Russell, subject: Nine Mi. Pt. 2 Augmented Inspection Team Findings. (4 pages)
52.	2/10/88	Memo from H. Kerch to W. Cook, subject: Closing Violation 50-410/87-25-01. (1 page)
53.	Undated	NMP-2 Enforcement Conference Briefing, with annotations. (2 pages)
54.	Undated	NMP-2 Enforcement Conference Briefing Package - 3/18/88. (3 pages)
55.	4/18/88	Pre-Inspection Cover Sheet and Inspection Plan. (1 page)
56.	4/20/88	Region I Morning Report. (1 page)

Re: FOIA-90-269

APPENDIX F
DOCUMENTS BEING PLACED IN THE PDR
(Continued)

NUMBER	DATE	DESCRIPTION
57.	6/2/88	Letter from R. Golden to G. Lavine, subject: Employees trapped in the Nine Mi. 2 Main Stear Tunnel on 9/14/97. (3 pages)
58.	8/2/88	Memo from H. Gregg to J. Johnson, subject: IST Allegations NMP Units 1 and 2. (2 pages)

Re: FOIA-90-269

APPENDIX G
DOCUMENTS BEING WITHHELD IN THEIR ENTIRETY

NUMBER	DATE	DESCRIPTION
1.	Undated	Post Inspection SALP Evaluation worksheets for Inspection Report No. 84-10. (3 pages) Exemption 5
2.	Undated	Post Inspection SALP Evaluation worksheets for Inspection Report No. 85-48. (3 pages) Exemption 5
3.	2/25/87	SALP Physical Protection - draft safeguards section input to SALP report for assessment period 2/1/86 - 1/31/87. (6 pages) Exemption 5
4.	2/24/86	SALP Physical Protection - draft safeguards section input to SALP report for assessment period 2/1/85 - 1/31/86. (5 pages) Exemption 5
5.	2/26/88	SALP Physical Protection - draft safeguards section input to SALP report for assessment period 10/31/86 - 2/14/88. (5 pages) Exemption 5
6.	4/7/88	Memo from W. T. Russell to James Lieberman, subject: Proposed Letter to Niagara Mohawk Power Corporation (NMPC) Based on Findings of An OI Investigation At Nine Mile Point, Unit 2. (1 page) Exemption 5

Re: FOIA-90-269

APPENDIX H
DOCUMENTS BEING WITHHELD IN PART

NUMBER	DATE	DESCRIPTION
1.	4/15/85	Letter from A. Schwencer to J. Sunser, subject: Nine Mile Point Unit 2 Physical Security Plan. (3 pages) Portions withheld pursuant to Exemptions 3 and 4.
2.	11/10/86	Letter from Joseph Beratta to Richard Starostecki, regarding Report of Physical Security event. (4 pages) Portions withheld pursuant to Exemption 3.
3.	1/30/87	Letter from Joseph Beratta to Richard Starostecki, regarding Report of Physical Security Event. (4 pages) Portions withheld pursuant to Exemption 4.
4.	5/29/87	Letter from Joseph Beratta to T. Murley, regarding Report of Physical Security Event. (4 pages) Portions withheld pursuant to Exemptions 3 and 4.
5.	8/3/87	Letter from Joseph Beratta to Thomas Murley, regarding Report of Physical Security Event. (3 pages) Portions withheld pursuant to Exemption 3.
6.	8/12/87	Letter from Thomas Martin to Joseph Beratta, subject: Combined Inspection Report Nos. 50-220/86-23 and 50-410/86-59. (10 pages) Portions withheld pursuant to Exemption 3.
7.	9/16/87	Letter from Joseph Beratta to Thomas Murley, regarding Report of Physical Security Event. (3 pages) Portions withheld pursuant to Exemptions 3 and 4.
8.	11/18/88	Letter from Stewart Ebneter to C. V. Mangan, subject: Combined Inspection No. 50-220/88-30 and 50-410/88-29. (12 pages) Portions withheld pursuant to Exemptions 3 and 4.

FOIA - 90-269

RESPONSE TYPE
XX FINAL [ ] PARTIAL

DATE DEC 1 3 1990

DOCKET NUMBER(S) (If applicable)

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST

REQUESTER

John C. Person

PART I. - AGENCY RECORDS RELEASED OR NOT LOCATED (See checked boxes)

No agency records subject to the request have been located.

No additional agency records subject to the request have been located.

Requested records are available through another public distribution program. See Comments Section.

XX Agency records subject to the request that are identified on Appendix(es) I are already available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC 20555.

XX Agency records subject to the request that are identified on Appendix(es) J are being made available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.

The nonproprietary version of the proposal(s) that you agreed to accept in a telephone conversation with a member of my staff is now being made available for public inspection and copying at the NRC Public Document Room, 2120 L Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.

Agency records subject to the request that are identified on Appendix(es) ______ may be inspected and copied at the NRC Local Public Document Room identified in the Comments Section.

Enclosed is information on how you may obtain access to and the charges for copying records placed in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC.

Agency records subject to the request are enclosed.

Records subject to the request have been referred to another Federal agency(ies) for review and direct response to you.

You will be billed by the NRC for fees totaling $ ______

In view of NRC's response to this request, no further action is being taken on appeal letter dated ______, No. ______

PART II. A - INFORMATION WITHHELD FROM PUBLIC DISCLOSURE

X Certain information in the requested records is being withheld from public disclosure pursuant to the exemptions described in and for the reasons stated in Part II, sections B, C, and D. Any released portions of the documents for which only part of the record is being withheld are being made available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC in a folder under this FOIA number and requester name.

COMMENTS

The combined actual fees for processing your two FOIA requests, FOIA-90-269 and FOIA-90-270, are as follows:

Search:	$ 94.24	[8 hrs clerical @ 11.78 pr hr]
	1009.55	[40 hrs, 35 mins professional @ 24.88 pr hr]
	89.99	[2 hrs, 15 mins. SES @ 39.97 pr hr]
Review:	58.90	[5 hrs clerical @ 11.78 pr hr]
	1073.94	[43 hrs, 10 mins professional @ 24.88 pr hr]
	$2326.62	TOTAL
	$6078.64	[combined estimated fees paid 7/17/90]
	-2326.62	[combined actual fees]
	$3752.02	REFUND

You will receive a refund from the NRC Division of Accounting in the amount of $3752.02.

SIGNATURE, DIRECTOR, DIVISION OF FREEDOM OF INFORMATION AND PUBLICATIONS SERVICES

Donnie H. Grimsley

9102080279

NRC FORM 464

PART II.B - APPLICABLE EXEMPTIONS

Records subject to the request that are described on the enclosed Appendix(es) K are being withheld in their entirety or in part under the Exemptions and for the reasons set forth below pursuant to 5 U.S.C. 552(b) and 10 CFR 9.17(a) of NRC Regulations.

1. The withheld information is properly classified pursuant to Executive Order. (EXEMPTION 1)

2. The withheld information relates solely to the internal personnel rules and procedures of NRC. (EXEMPTION 2)

X 3. The withheld information is specifically exempted from public disclosure by statute indicated. (EXEMPTION 3)

Sections 141-145 of the Atomic Energy Act which prohibits the disclosure of Restricted Data or Formerly Restricted Data (42 U.S.C. 2161-2165).

X Section 147 of the Atomic Energy Act which prohibits the disclosure of Unclassified Safeguards Information (42 U.S.C. 2167).

X 4. The withheld information is a trade secret or commercial or financial information that is being withheld for the reason(s) indicated. (EXEMPTION 4)

The information is considered to be confidential business (proprietary) information.

X The information is considered to be proprietary information pursuant to 10 CFR 2.790(d)(1).

The information was submitted and received in confidence pursuant to 10 CFR 2.790(d)(2).

X 5. The withheld information consists of interagency or intraagency records that are not available through discovery during litigation. (EXEMPTION 5) Applicable Privilege:

X Deliberative Process: Disclosure of predecisional information would tend to inhibit the open and frank exchange of ideas essential to the deliberative process. Where records are withheld in their entirety, the facts are inextricably intertwined with the predecisional information. There also are no reasonably segregable factual portions because the release of the facts would permit an indirect inquiry into the predecisional process of the agency.

Attorney work-product privilege. (Documents prepared by an attorney in contemplation of litigation.)

Attorney-client privilege. (Confidential communications between an attorney and his/her client.)

6. The withheld information is exempted from public disclosure because its disclosure would result in a clearly unwarranted invasion of personal privacy. (EXEMPTION 6)

7. The withheld information consists of records compiled for law enforcement purposes and is being withheld for the reason(s) indicated. (EXEMPTION 7)

Disclosure could reasonably be expected to interfere with an enforcement proceeding because it could reveal the scope, direction, and focus of enforcement efforts, and thus could possibly allow them to take action to shield potential wrongdoing or a violation of NRC requirements from investigators. (EXEMPTION 7(A))

Disclosure would constitute an unwarranted invasion of personal privacy. (EXEMPTION 7(C))

The information consists of names of individuals and other information the disclosure of which could reasonably be expected to reveal identities of confidential sources. (EXEMPTION 7(D))

OTHER

PART II.C - DENYING OFFICIALS

Pursuant to 10 CFR 9.25(b) and/or 9.25(c) of the U.S. Nuclear Regulatory Commission regulations, it has been determined that the information withheld is exempt from production or disclosure, and that its production or disclosure is contrary to the public interest. The persons responsible for the denial are those officials identified below as denying officials and the Director, Division of Freedom of Information and Publications Services, Office of Administration and Resources Management, for any denials that may be appealed to the Executive Director for Operations (EDO).

DENYING OFFICIAL	TITLE/OFFICE	RECORDS DENIED	APPELLATE OFFICIAL	
			SECRETARY	EDO
Thomas E. Murley	Director, NRR	K/1, K/3, K/7, K/8		X
Thomas T. Martin	Regional Administrator, Reg. I	K/2, K/4, K/6		X
John C. Hoyle	Assistant Secretary of the Commission	K/5	X	

PART II.D - APPEAL RIGHTS

The denial by each denying official identified in Part II.C may be appealed to the Appellate Official identified in that section. Any such appeal must be in writing and must be made within 30 days of receipt of this response. Appeals must be addressed as appropriate to the Executive Director for Operations or to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should clearly state on the envelope and in the letter that it is an "Appeal from an Initial FOIA Decision."

Re: FOIA-90-269

APPENDIX I
DOCUMENTS ALREADY AVAILABLE IN THE PDR

NUMBER	DATE	DESCRIPTION
1.	3/13/89	Letter from William T. Russell to Lawrence Burkhardt, subject: Notice of Violation (NRC Inspection Report No. 50-410/88-21). PDR Accession No. 8903230219
2.	3/13/89	Notice of Violation (EA 89-04). PDR Accession No. 8903230221

Re: FOIA-90-269

APPENDIX J
DOCUMENTS BEING PLACED IN THE PDR

NUMBER	DATE	DESCRIPTION
1.	10/4/83	Memo from J. Taylor to T. Murley, subject: Construction Appraisal Inspection - Nine Mile Point Unit 2 - Docket No. 50-410. (5 pages)
2.	3/12/84	Memo from J. Nelson Grace to Richard Starostecki, subject: Nine Mile Point Unit 2 CAT Inspection. (5 pages)
3.	2/5/86	Memo from G. McCorkle to J. Joyner, subject: SALP Input - Nine Mile Point Unit 2 (February 1, 1985 - January 31, 1986). (2 pages)
4.	10/9/86	Meeting Summary for September 24, 1986, Meeting on Main Steam Isolation Valves - Nine Mile Point, Unit 2. (3 pages)
5.	Undated	Handwritten notes of 10/15/86 meeting on Nine Mile Point, Unit 2, attaching various view-graphs and diagrams. (40 pages)

Re: FOIA-90-269

APPENDIX K
DOCUMENTS BEING RELEASED IN PART

NUMBER	DATE	DESCRIPTION
1.	10/28/83	Memo from Mary Haughey to Darrell Eisenhut, subject: NRR Input to SALP - Nine Mile Point 2. (3 pages) Portions withheld pursuant to Exemption 5
2.	1/15/86	Letter from Thomas T. Martin to Joseph Sunser, subject: Inspection No. 50-410/85-38. (14 pages) Portions withheld pursuant to Exemptions 3 and 4.
3.	2/23/86	Memo from Mary Haughey to Richard Starostecki, subject: NRR SALP Input -Nine Mile Point Nuclear Station Unit 2. (12 pages) Portions withheld pursuant to Exemption 5.
4.	2/5/88	Letter from Thomas Martin to Joseph P Beratta, subject: Combined Inspection No. 50-220/87-14 and 50-410/87-30. (6 pages) Portions withheld pursuant to Exemptions 3 and 4.
5.	5/20/88	SECY-88-137 - Proposed Letter to Niagara Mohawk Power Corporation (NMPC) Based on the Submittal of Incomplete and Misleading Information to the NRC Concerning An Internal Investigation Performed by the Company at Nine Mile Point, Unit 2. (5 pages released) Portions withheld pursuant to Exemption 5.
6.	2/21/89	Memo from William T. Russell to James Lieberman, subject: Proposed Enforcement Action - Niagara Mohawk Power Corporation (Nine Mile Point, Unit 2). (1 page released) Portions withheld pursuant to Exemption 5.

Re: FOIA-90-269

APPENDIX K
DOCUMENTS BEING RELEASED IN PART
(Continued)

NUMBER	DATE	DESCRIPTION
7.	2/28/89	Memo from Steven A. Varga to James Lieberman, subject: Proposed Civil Penalty - Nine Mile Point 2. (1 page released) Portions withheld pursuant to Exemption 5.
8.	6/11/85	Memo from G. McCorkle to Cecil Thomas, subject: Niagara Mohawk Power Corporation Nine Mile Point Nuclear Station Unit 2 Safety Evaluation Report. (4 pages) Portions withheld pursuant to Exemption 3.

NEWMAN & HOLTZINGER, P.C.

1615 L STREET, N.W.
WASHINGTON, D.C. 20036

202 955 6600

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ALVIN H. GUTTERMAN
EDWARD J. TWOMEY
JAMES B. WILCOX JR
KEVIN P. GALLEN
THOMAS A. SCHMUTZ
MICHAEL F. HEALY
ROBERT I. WHITE
SCOTT A. HARMAN
STEVEN P. FRANTZ
DAVID B. RASKIN
KEVIN J. LIPSON
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CHARLES C. THEBAUD JR
NANCY A. WHITE

ROBERT LOWENSTEIN
HERBERT B. COHN
JAY H. GUTIERREZ*
KENNETH M. KASTNER
SCOTT SLAUGHTER
OF COUNSEL

*NOT ADMITTED IN D.C.

June 13, 1990

HAND DELIVERED

FREEDOM OF INFORMATION
ACT REQUEST
FOIA-90-269
Rec'd 6-13-90

Director, Division of Freedom of
Information and Publication
Services
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Ms. Linda Robinson

Re: Freedom of Information Act Request

Dear Ms. Robinson:

Pursuant to the Freedom of Information Act (FOIA), 5 U.S.C. § 552, *et seq.*, as supplemented by the NRC's implementing regulations, 10 C.F.R. § 9.11, *et seq.*, we hereby request that you produce for inspection and copying the documents described in the attachments to this letter. This request primarily involves documents relating to site inspections by NRC inspectors conducted during the construction of Nine Mile Point Unit 2 (NMP-2), NRC Docket No. 50-410. Set forth in Attachment A hereto is a listing of the particular inspections for which we seek to review related documents. For each such inspection, we would like to review the following documents:

1. All documents that relate or pertain to the selection of the inspection team, including the selection of any consultants or third parties to serve as team members.

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29pp

2. All documents relating to the reason or reasons for conducting the inspection.

3. All documents setting forth the goals and objectives of the inspection.

4. All documents setting forth the guidelines, rules, or procedures to be followed in the inspection.

5. All documents that pertain or relate to the selection of functional areas subject to evaluation and the actual measurement of effectiveness in these areas.

6. All documents that relate or pertain to the criterion by which the overall conclusions or results of the inspection were measured.

7. All reports, evaluations, or analyses prepared by any consultant or third party in connection with the inspection.

8. All correspondence, memoranda, and other communications between the offices and divisions of the NRC, including communications to and from Region I and the Resident Inspector at NMP-2, relating or pertaining to the inspection.

9. All intra-divisional and intra-office correspondence and memoranda concerning the inspection.

10. All personal files, memoranda, and notes of each inspector identified as participating in the inspection for each inspection listed in Attachment A.

11. All personal files, memoranda, and notes of those persons identified as approving the inspection report for each inspection listed in Attachment A.

As used herein, the term "NRC" includes all offices and divisions within the Agency having any involvement with the subject matter of this request, including the NRC Regional Office for Region I, the NRC Resident Inspector's office at the NMP-2 site, and the Office of Inspection and Enforcement. As also used herein, the term "document", unless otherwise specifically limited, means all correspondence, letters, memoranda (internal and external), records of telephone conversations, notes, reports, agreements, guidelines, procedures, meeting slides, and the like, whether in draft or final form, that are in any way relevant to the specific document descriptions.

We look forward to your response within the time limits prescribed by the regulations. We would note that in the event you consider any documents to be exempt from production, the non-exempt portions should be released. Again, we are willing to provide you with any additional information or offer any clarification you may need in processing this request. Further, given that this FOIA request is fairly extensive, we would ask that you consider releasing the documents to us in stages as you accumulate them rather than waiting until all documents subject to production have been accumulated.

As prescribed by the regulations, we agree to pay whatever charges are incurred in processing this request. Please feel free to call me at (202) 955-6790 if you have any questions concerning this matter.

Very truly yours,

John C. Person

JCP:lmd

Attachment A

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
1981			
1/20 - 1/23	81.01	A. V. Varela	S. D. Ebneter
2/18 & 2/25	81.02	T. J. Jackson	R. J. Bores
4/21 - 4/23	81.03	R. A. Feil	H. B. Kister
4/22	81.04	B. H. Grier E. J. Brunner H. B. Kister S. D. Hudson R. A. Feil	E. J. Brunner
6/23 - 6/25	81.05	R. A. Feil	H. B. Kister
7/14 - 7/16	81.06	R. A. Feil	W. Baunach for H. B. Kister
7/27 - 7/31	81.07	R. Paolino A. A. Varela	L. E. Tripp for S. D. Ebneter
8/4 - 8/6	81.08	R. A. Feil	H. B. Kister
8/18 - 8/21	81.09	W. F. Sanders R. A. McBreaty S. D. Reynolds	L. E. Tripp
9/1 - 9/3	81.10	R. A. Feil R. D. Schulz	H. B. Kister
9/29 - 9/30	81.11	R. A. Feil R. D. Schulz H. B. Kister	H. B. Kister
10/13 - 11/13	81.12	R. D. Schulz	H. B. Kister
11/30 - 12/18	81.13	S. K. Chaudhary R. J. Paolino S. D. Reynolds L. E. Tripp R. D. Schulz	S. D. Ebneter
12/21/ - 1/15/82	81.14	R. D. Schulz	H. B. Kister

NEWMAN & HOLTZINGER, P. C.

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
1982			
1/18 - 2/26	82.01	R. D. Schulz	H. B. Kister
3/1 - 3/26	82.02	R. D. Schulz	H. B. Kister
3/29 - 4/30	82.03	R. D. Schulz	H. B. Kister
5/11 - 5/13	82.04	A. E. Finkel	D. A. Beckman
5/10 - 6/3	82.05	R. D. Schulz	H. B. Kister
(Enforcement Conference) 6/1	82.06	R. W. Starostecki E. J. Brunner H. B. Kister R. D. Schulz	H. B. Kister
6/21 - 7/23	82.07	R. D.Schulz	H. B. Kister
7/16 (Enforcement Conference)	82.08	E. J. Brunner S. D. Ebneter D. J. Holody R. W. Starostecki R. D. Schulz	H. B. Kister
7/13 - 7/16 & 7/20	82.09	R. J. Paolino S. Richards	D. A. Beckman
7/26 - 8/27	82.10	R. D. Schulz	H. B. Kister
8/30 - 9/30	82.11	R. D. Schulz A. E. Finkel	H. B. Kister
10/12 - 11/12	82.12	R. D. Schulz	H. B. Kister
10/20 (Enforcement Conference)	82.13	R. D. Haynes D. J. Holody H. B. Kister R. B. Schulz R. W. Starostecki	H. B. Kister
11/15 - 12/22	82.14	R. D. Schultz	H. B. Kister
12/13 - 12/17	82.15	A. A. Varela R. D. Schulz	J. P. Durr
12/14 - 12/16	82.16	A. E. Finkel	C. Anderson

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
1983			
1/13 - 2/4	83.01	R. D. Schulz J. Grant	H. B. Kister
2/7 - 3/11	83.02	R. D. Schulz	H. B. Kister
3/1 - 3/3	83.03	A. Finkel	C. Anderson
3/14 - 4/15	83.04	R. D. Schulz	H. B. Kister
4/25 - 5/27	83.05	R. D. Schulz R. A. Gramm	H. B. Kister
5/16 - 6/1	83.06	L. R. Plisco R. D. Schulz	C. Anderson
6/13 - 6/17 & 7/13 - 8/5	83.07	R. A. Gramm	R. M. Gallo
6/7 - 6/9	83.08	A. Finkel	C. Anderson
5/23 (Management Meeting)	83.09	S. D. Ebneter R. R. Keimeg H. B. Kister R. W. Starostecki	R. R. Keimeg
7/25 - 7/29	83.10	L. Narrow E. H. Gray	J. P. Durr
8/? - 8/4	83.11	A. Finkel R. Gramm	C. Anderson
8/8 - 9/21	83.12	R. Gramm J. Grant	R. M. Gallo
9/14	83.13	R. K. Struckmeyer R. T. Hogan	W. J. Pasciak
8/30 (Enforcement Conference)	83.14	J. A. Allan S. D. Ebneter R. M. Gallo R. A. Gramm J. M. Grant E. H. Gray D. J. Holody H. B. Kister T. Martin L. M. Narrow R. W. Starostecki	R. Gallo

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
10/18 - 10/20	83.15	A. Finkel R. Gramm	C. J. Anderson
10/11 - 12/2	83.16	R. Gramm	S. Collins
12/15 - & 1/20/84	83.17	R. Gramm W. J. Lazarus	S. Collins
11/7 - 19 & 11/28 - 12/9	83.18	A. B. Beach G. B. Georgiev W. A. Hanson D. B. Osborne H. W. Philips H. J. Wong J. M. Grant Also Consultants: R. M. Compton D. C. Ford W. S. Marini F. Y. Martindale F. A. Pimentel	R. F. Heishman

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
1984			
1/23 - 3/2	84.01	R. A. Gramm D. Terao	S. J. Collins
2/7 - 2/9	84.02	A. Finkel	C. J. Anderson
2/22	84.03	R. Gramm J. Allan J. Axelrad S. J. Collins J. Craig R. DeYoung S. Ebneter R. A. Gramm J. Grant J. Gutierrez D. Holody S. Hudson H. B. Kister G. Kliner W. Lazarus J. Lieberman T. Martin T. Murley R. Starostecki	S. J. Collins
4/9 - 5/11	84.06	R. A. Gramm	S. J. Collins
5/14 - 5/18	84.07	L. Narrow	J. P. Durr
4/30 - 5/25	84.08	H. W. Kerch R. H. Harris R. M. Campbell	J. P. Durr
5/14 - 6/15	84.09	R. A. Gramm S. K. Chaudhary	S. J. Collins
5/21 - 5/24	84.10	R. J. Bailey J. M. Dunlap	R. R. Keimog
6/18 - 7/27	84.11	R. A. Gramm A. C. Cerne J. M. Grant	S. J. Collins
7/30 - 9/6	84.13	R. A. Gramm	S. J. Collins
9/10 - 11/2	84.15	R. A. Gramm	S. J. Collins
10/29 - 11/2	84._6	R. L. Nimitz	W. J. Pasciak

NEWMAN & HOLTZINGER, P. C.

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
10/29 - 11/2	84.17	L. Narrow	J. P. Durr
12/3 - 12/14	84.18	J. P. Durr A. E. Finkel R. A. Gramm R. H. Harris H. W. Kerch K. A. Manoly G. Napuda J. H. Raval S. D. Reynolds	S. D Ebneter
11/5 - 12/21	84.19	R. A. Gramm R. M. Wheeler	W. J. Lazarus
11/14	84.20	R. A. Gramm S. D. Hudson	W. J. Lazarus
12/24 - 2/1/85	84.21	R. A. Gramm R. M. Wheeler	W. J. Lazarus

C

Newman & Holtzinger, P. C.

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
1985			
1/15 - 1/18	85.01	L. Briggs S. Hudson R. Gramm R. Wheeler	L. Bettenhausen
1/28 - 2/1	85.02	S. D. Reynolds R. Gramm R. Wheeler	J. Durr
2/11 - 2/15	85.03	R. Paolino C. Woodard A. Varela R. Gramm	C. Anderson
2/4 - 3/18	85.04	R. A. Gramm R. M. Wheeler	J. C. Linville
2/6	85.05	R. A. Gramm	W. J. Lazarus
3/4 - 3/8	85.06	G. Napuda K. A. Manoly H. Van Kessel R. Gramm S. Hudson R. Wheeler	P. K. Eapen
3/18 - 3/22	85.08	A. Finkel L. Cheung C. Woodard S. Ebneter R. Gramm	C. J. Anderson
2/15 & 3/15	85.09	R. A. Gramm J. C. Linville S. C. Ebneter S. J. Collins	J. C. Linville
3/19 - 4/26	85.10	R. A. Gramm S. D. Hudson R. M. Wheeler A. J. Luptak H. W. Kerch	J. C. Linville

Newman & Holtzinger, P. C.

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
4/2 - 4/4 & 4/9 - 4/12	85.11	H. I. Gregg A. Kortas R. Gramm S. Hudson J. Linville R. Wheeler A. Luptak	J. T. Wiggins
5/6 - 5/10	85.12	A. Finkel	C. J. Anderson
4/29 - 6/7	85.13	R. A. Gramm G. A. Walton R. M. Wheeler	J. C. Linville
6/11 - 6/19	85.15	R. Keller D. Lange F. Crescenzo J. Berry W. Cliff G. Sly	H. Kister
5/28 - 5/31	85.16	R. A. McBrearty R. A. Gramm R. M. Wheeler	J. T. Wiggins
6/10 - 6/14	85.17	F. Paulitz R. A. Gramm R. Wheeler	C. J. Anderson
5/21 - 5/24	85.18	G. S. Lewis E. V. Imbro J. L. Milhoan G. T. Ankrum S. C. Ebneter B. K. Grimes	J. L. Milhoan
6/10 - 7/19	85.19	R. A. Gramm R. M. Wheeler	J. C. Linville
6/24 - 6/28	85.20	R. L. Nimitz M. J. Cioffi L. E. Meyers S. Hudson R. Gramm	W. J. Pasciak

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
6/17 (Management Conference)	85.21	S. J. Collins J. P. Durr S. D. Ebneter R. A. Gramm J. Linville K. Manoly T. Murley R. W. Starostecki J. Wiggins H. J. Wong	J. C. Linville
7/16 - 7/19	85.23	R. A. McBrearty R. A. Gramm	J. T. Wiggins
7/22 - 7/26	85.24	A. A. Varela R. A. Gramm	J. T. Wiggins
7/22 - 8/30	85.25	R. A. Gramm	J. C. Linville
7/23 (Management Conference)	85.26	J. P. Durr R. W. Eselgroth R. A. Gramm J. Linville T. Murley R. W. Starostecki	J. C. Linville
9/9 - 1/18/86	85.27	L. T. Doerflein R. A. Gramm J. M. Grant S. D. Hudson J. P. Rogers	J. C. Linville
8/12 - 8/16	85.28	G. S. Lewis E. V. Imbro J. L. Milhoan G. T. Ankrum S. E. Ebneter B. K. Grimes	J. L. Milhoan
9/30 - 10/4	85.29	E. H. Gray	J. T. Wiggins
9/9 - 9/13	85.30	L. Briggs L. Doerflein R. Gramm S. Hudson	P. Eselgroth
10/7 - 10/11	85.31	K. A. Manoly R. M. Campbell R. A. Gramm	J. T. Wiggins

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
10/14 - 10/18	85.32	R. L. Nimitz S. Hudson R. Gramm	W. J. Pasciak
10/21 - 10/25	85.33	H. I. Gregg R. Gramm	J. T. Wiggins
10/21 - 10/25	85.34	A. G. Krasopoulos D. Kubicki A. Coppola K. Parkinson R. Gramm	C. J. Anderson
10/28 - 11/1	85.35	C. H. Woodard L. Doerflein	G. Anderson
10/21 - 11/26	85.36	R. A. Gramm L. T. Doerflein S. D. Hudson	J. C. Linville
10/28 - 11/1	85.37	S. D. Hudson G. S. Marshall P. H. Bissett F. J. Crescenzo D. J. Lange	J. C. Linville
5/21 - 5/24	85.38	J. M. Dunlap R. R. Keimeg Joyner Martin	R. R. Keimeg
11/25 - 11/27	85.40	A. Della Rattan R. R. Keimeg Joyner Martin	R. R. Keimeg
12/10 - 12/19	85.41	D. Lange R. Keller F. Crescenzo A. Howe B. Hajek G. Sly W. Cliff L. Miller	H. Kister
11/12 - 11/15	85.42	R. J. Paolino R. A. Gramm	C. J. Anderson

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
11/25 - 12/5 & 12/9 - 12/19	85.43	H. W. Kerch R. H. Harris R. M. Campbell R. Gramm	J. T. Wiggins
12/2 - 1/10/86	85.44	L. T. Doerflein R. A. Gramm S. D. Hudson H. W. Kerch	J. C. Linville
12/9 - 12/10	85.45	A. Krasopoulos	C. Anderson
12/9 - 12/12	85.46	R. A. McBrearty	J. T. Wiggins
12/16 - 12/20	85.47	R. L. Nimitz S. Hudson R. Gramm	M. Shanbaky

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
1986			
1/13 - 2/21	86.01	L. T. Doerflein R. A. Gramm S. D. Hudson M. Miller R. Paolino	J. C. Linville
1/6 - 1/17	86.02	J. Linville R. Gramm S. Hudson L. Doerflein	Parameter, Inc.
1/21 - 1/24	86.03	L. Briggs	P. Eselgroth
1/27 - 1/31	86.04	J. C. Linville R. Paolino L. Derflein A. Cerne J. Isom	J. C. Linville
3/3 - 3/7	86.05	L. Briggs R. Gramm S. Hudson	P. Eselgroth
1/22	86.06	R. A. Gram	J. C. Linville
1/3 & 1/7 Project Audit #50 & C/A I.R.	86.07	NRC Audit Team (No names mentioned) Report prepared by Report approved by	 E. V. Imbro G. T. Ankrum
3/3 - 3/7	86.08	F. Paulitz L. Cheung R. Gramm S. Hudson	C. J. Anderson
2/22 - 4/18	86.09	R. A. Gramm S. D. Hudson G. W. Meyer J. R. Stair	J. C. Linville
3/31 - 4/4	86.11	G. Napuda J. Gilray W. Oliveira S. Hudson R. Gramm	P. K. Eapen

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
4/14 - 4/25	86.13	K. Manoly S. Chaudhary A. Lodewyk J. Hunter G. Woodard F. Paulitz R. A. Gramm M. V. Johnston	J. Wiggins
3/31 - 4/4	86.14	M. Evans	P. Eselgroth
4/7 - 4/11 & 4/21 - 4/25	86.15	C. Petrone L. Briggs M. Dev R. Gramm S. Kucharski S. Hudson	J. Johnson
4/7 - 4/15	86.16	S. Kucharski R. Gramm S. Hudson C. Petrone	C. Anderson
4/28 - 5/1	86.17	J. J. Kottan M. E. Kramaric	W. J. Pasciak
4/19 - 5/31	86.18	W. A. Cook R. A. Gramm H. W. Kerch A. J. Lodewyk A. J. Luptak C. S. Marschall G. W. Meyer J. R. Stair	J. C. Linville
5/19	86.19	R. L. Nimitz M. Kaminski	M. Shanbaky
5/5 - 5/9	86.20	L. Briggs D. Florek R. Gramm J. Hunter J. Stair	P. Eselgroth
5/12 - 5/15	86.21	A. Krasopoulos H. Kerch R. Gramm J. Stair C. Marshall	C. Anderson

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
4/30 - 5/9	86.22	R. W. Winters J. G. Hunter R. A. Gramm J. Stair	P. K. Eapen
5/19 - 5/20	86.23	C. Gordon B. Haagensen C. Hawley A. Smith	W. Lazarus
5/12 - 5/16	86.24	W. G. Martin G. C. Smith	R. R. Keimeg
5/20 - 5/23	86.26	C. Petrone A. Finkel R. Gramm	J. Johnson
5/19 - 5/23	86.27	M. Evans R. Gramm A. Finkel C. Petrone	P. Eselgroth
5/27 - 6/13	86.28	C. H. Woodard H. I. Gregg W. J. Butler R. A. Gramm	C. J. Anderson
6/1 - 7/13	86.29	W. A. Cook R. A. Gramm J. C. Linville G. W. Meyer J. R. Stair	J. C. Linville
6/2 - 6/6	86.30	L. Briggs W. Cook R. Gramm J. Stair	P. Eselgroth
6/16 - 6/27	86.31	L. Briggs M. Evans R. Brady R. Gramm	P. Eselgroth
6/23 - 6/25	86.32	E. H. Gray	J. T. Wiggins
6/30 - 7/11	86.33	L. Briggs M. Evans W. Cook R. Gramm	P. Eselgroth

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
7/7 - 7/11	86.35	J. J. Kottan K. K. Rabatin	W. J. Pasciak
7/7 - 7/11	86.36	J. A. Prell M. Evans J. Kottan K. Rabatin	J. Johnson
7/14 - 7/18 7/28 - 8/1 8/11 - 8/15	86.37	T. Koshy C. Woodward J. Paolino L. S. Cheung	C. J. Anderson
7/14 - 7/24	86.38	D. Florek M. Evans E. Vanterpool W. Cook	P. Eselgroth
7/7 - 8/31	86.39	W. A. Cook R. A. Gramm C. S. Marschall G. W. Meyer W. L. Schmidt	J. C. Linville
7/28 - 8/1	86.40	W. G. Martin	R. R. Keimeg
7/29 - 8/1	86.41	D. Florek J. Golla W. Cook T. Koshy R. McBrearty W. Schmidt C. Woodard	P. Eselgroth
9/1 - 9/30	86.42	W. A. Cook C. S. Marschall G. W. Meyer R. L. Nimitz W. L. Schmidt	J. C. Linville
7/28 - 8/1	86.43	R. A. McBrearty W. Cook W. Schmidt	J. T. Wiggins
8/4 - 8/6	86.44	W. Thomas C. Conklin	W. Lazarus
7/21 - 7/25	86.45	R. Struckmeyer M. Krammaric	W. Pasciak

NEWMAN & HOLTZINGER, P. C.

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
8/4 - 8/7	86.46	R. L. Nimitz S. Sherbini	M. Shanbak
8/4 - 8/8	86.47	S. K. Chaudhary W. A. Cook	J. Johnson
8/11 - 8/15	86.48	W. Oliveri R. W. Winters	Dr. P. K. Eapen
8/18 - 8/22	86.49	R. K. Struckmeyer M. E. Kramaric	W. J. Pasciak
8/18 - 8/28	86.50	L. Briggs D. Florek M. Evans	P. Eselgroth
9/8 - 9/19	86.51	D. Florek M. Evans	L. Briggs
8/26 - 8/28 & 9/8 - 9/12	86.52	J. C. Linville Dr. P. K. Eapen R. Gramm J. Stair	R. M. Gallo
9/22 - 9/26	86.53	H. I. Gregg	J. R.Strosnider
9/20 - 10/2	86.54	J. Hawxhurst	W. Lazarus
10/7 - 10/9	86.55	H. Zibulsky	W. J. Pasciak
10/1 - 11/16	86.56	W. A. Cook P. K. Eapen J. E. Kaucher C. S. Marschall W. L. Schmidt C. H. Woodard	J. C. Linville
10/20 - 10/24	86.57	D. Florek M. Evans	L. Briggs
10/27 - 10/29	86.58	J. Hawxhurst W. Cook S. Merwin M. Moeller F. Victor M. Clausen J. Kaucher W. Schmidt C. Marshall	W. Lazarus

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
10/31 - 11/5	86.60	M. Evans	D. Florek
8/25 - 8/29	86.61	S. Kucharski E. Kelly C. Marschall G. Napuda R. Paolino W. Raymond R. Mutakas S. Collins	W. Kane
11/12	86.62	J. C. Linville Eselgroth S. C. Collins Kane Allan Morley	
11/17 - 11/21	86.63	R. L. Nimitz	M. M. Shanbaky
11/17 - 11/21	86.64	M. Evans	D. Florek
11/17 - 1/4/87	86.65	W. A. Cook C. S. Marschall G. W. Meyer W. L. Schmidt J. C. Linville	J. E. Kaucher
11/17 - 1/4/87	86.66	W. A. Cook J. E. Kaucher C. S. Marschall G. W. Mayer W. L. Schmidt	J. C. Linville
12/1 -12/5	86.67	H. I. Gregg W. Cook W. Schmidt	J. R. Strosnider
12/8 - 12/12	86.68	M. Evans W. Cook W. Schmidt	C. Petrone

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
1987			
1/4 - 3/1	87.02	W. A. Cook C. S. Marschall W. L. Schmidt G. Mayer	J. C. Linville
1/12 - 1/28	87.03	H. I. Gregg	J. R. Strosnider
1/12 - 1/16	87.04	D. LeQuia W. Cook C. Marschall W. Schmidt	M. Shanbaky
2/2 - 2/5	87.05	F. Paulitz W. Cook W. Schmidt C. Marschall	C. J. Anderson
2/10 - 2/13	87.06	M. Evans L. Wink W. Cook C. Marschall W. Schidt	P. Eselgroth
2/9 - 2/13	87.07	A. Finkel W. Cook C. Marschall W. Schmidt	N. Blumberg
4/20 - 6/7	87.08	W. A. Cook C. S. Marshall R. L. Nimitz W. L. Schmidt	J. R. Johnson
3/2 - 4/19	87.09	W. H. Bateman W. A. Cook D. J. Lange C. S. Marschall W. L. Schmidt	J. R. Johnson
4/6 - 4/10	87.10	W. G. Martin	R. I. Keimeg
3/23 - 3/27	87.11	H. I. Gregg W. Cook	J. R. Strosnider

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
3/23 - 3/27	87.12	L. Cheung C. Marschall W. Schmidt	C. J. Anderson
4/20 - 4/21	87.13	A. Krasopoulos	C. J. Anderson
4/13 - 4/17	87.14	H. I. Gregg	J. R. Strosnider
5/11 - 5/15	87.15	L. J. Wink W. Cook W. Schmidt	D. Florek
6/1 - 6/12	87.16	M. McBride C. Warren A. Luptak L. Wink D. Lange N. Perry M. Fairtile D. Beckman B. Jorgensen R. Nimitz J. Johnson	R. Gallo
5/19 - 5/29	87.17	M. Evans D. Florek W. Cook C. Marschall W. Schmidt	P. Eselgroth
6/10 - 6/11	87.19	W. Martin W. Lancaster	R. Keimeg
6/8 - 7/19	87.20	W. Cook C. Marschall W. Schmidt N. Perry G. Meyor R. Nimitz	J. R. Johnson
6/8 - 6/19	87.21	M. Evans L. Wink W. Cook C. Marschall W. Schmidt	P. Eselgroth

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
6/22 - 6/26 & 6/30 - 7/1	87.22	R. Nimitz H. Bicehouse C. Woodard W. Cook W. Schmidt C. Marschall	M. Shanbaky
6/22 - 6/30	87.23	M. Evans L. Wink W. Cook N. Perry W. Schmidt	P. Eselgroth
8/3 - 8/7	87.24	R. Stockmeyer A. Kirkwood	W. Pasciak
6/29 - 6/30 7/6 - 7/10	87.25	H. Kerch R. Gramm B. Cook	E. Gray
7/6 -7/10	87.26	M. Evans W. Cook C. Marschall W. Schmidt	P. Eselgroth
8/3 - 8/12	87.27	L. Wink D. Florek W. Cook C. Marschall N. Perry W. Schmidt	P. Eselgroth
7/20 -7/24	87.28	L. Wink W. Cook C. Marschall W. Schmidt	P. Eselgroth
7/20 - 8/30	87.29	W. Cook C. Marschall W. Schmidt N. Perry	J. Johnson
7/27 - 7/31	87.30	W. Martin W. Cook	R. Keimeg

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
8/25 - 8/27	87.31	T. Tuccinaret	W. Lazarus
		B. Fox	
		C. Conklin	
		W. Thomas	
		G. Stoetzel	
		W. Cook	
		C. Marschall	
		W. Schmidt	
9/1 - 9/8	87.32	W. Cook	J. Johnson
		C. Marschall	
		W. Schmidt	
8/24 - 8/28	87.33	W. Cook	D. Florek
		C. Marschall	
		W. Schimdt	
8/24 - 8/27	87.34	R. Nimitz	M. Shanbaky
		M. Markley	
		C. Marschall	
9/21 - 9/23 & 10/5 - 10/9	87.35	L. Wink	D. Lange
		W. Cook	
		C. Marschall	
		W. Schmidt	
9/14 - 9/18	87.36	B. Davidson	W. Pasciak
		W. Schmidt	
8/31 - 10/4	87.37	W. Cook	J. Johnson
		C. Marschall	
		W. Schmidt	
10/12 - 10/15	87.38	L. Wink	D. Lange
		W. Cook	
		C. Marschall	
		W. Schmidt	
10/5 - 10/30	87.39	W. Cook	J. Johnson
		C. Marschall	
		W. Schmidt	
		H. Kerch	
10/19 - 10/22	87.40	N. Perry	J. Johnson
		G. Meyer	
		W. Cook	
		W. Schmidt	

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
11/2 - 11/6	87.41	L. Wink G. Meyer W. Schmidt	D. Lange
10/31 - 12/10	87.42	W. Cook W. Schmidt G. Meyer E. Gray	J. Johnson
12/11	87.45	W. Cook C. Marschall W. Schmidt D. Florek A. Krasopoulos T. Lumb G. Meyer D. Persinko	J. Johnson

Newman & Holtzinger, P. C.

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
[illegible]			
1/21 - 1/24/	88.01	E. Wenzinger	E. Wenzinger
		M. Haughey	
		A. Howe	
		L. Lois	
		H. Ornstein	
2/1 - 3/31	88.02	W. Cook	J. Johnson
		A. Krasopoulos	
		R. Laura	
		G. Meyer	
		R. Plasse	
		W. Schmidt	
1/25 - 1/29	88.03	T. Lumb	D. Lange
		W. Cook	
		W. Schmidt	
2/24 - 2/26	88.04	D. Florek	D. Lange
		T. Lumb	
		C. Sisco	
		W. Cook	
2/15 - 2/19	88.05	R. Nimitz	N. Shanbaky
		M. Cook	
2/29 - 3/4 & 3/21 - 3/25	88.06	M. Dev	R. Gallo
		R. Temps	
		W. Oliveri	
		L. Prividy	
		T. Rebelowski	
		G. Napuda	
		W. Cook	
		R. Gallo	
		K. Hook	
		J. Johnson	
		R. Laura	
		W. Schmidt	
4/1 - 5/5	88.07	W. Cook	J. Johnson
		A. Krasopoulos	
		R. Laura	
		W. Schmidt	
4/18 - 4/22	88.08	L. Cheung	C. Anderson
		S. Alexander	
		E. Claiborne	
		W. Carpenter	
		G. Decker	

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
3/7 - 3/11 & 4/4 - 4/8	88.09	R. A. McBrearty J. Strosnider H. Kerch R. Harris M. Oliveri W. Cook W. Schmidt	J. R. Strosnider
4/4 - 4/8	88.10	W. Oliveri W. Cook	N. Blumberg
4/11 - 4/15	88.11	R. Paolino C. Anderson R. Benedict L. Cheung D. Caphton W. Cook J. Durr M. Haughey R. Gallo F. Hawkins J. Johnson P. Kelley	C. Anderson
4/18 - 4/22	88.12	R. Loesch R. Nimitz W. Cook	M. Shanbaky
4/25 - 4/29	88.13	A. Kirkwood J. Cottan	W. Pasciak
6/6 - 6/10	88.14	M. Evans W. Cook W. Schmidt	D. Lange
5/6 - 5/24	88.15	W. Cook W. Schmidt A. Krasopoulos	J. Johnson
5/25 - 7/6	88.16	W. Cook W. Schmidt R. Temps A. Krasopoulos	J. Johnson

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
7/7 - 8/24	88.17	W. A. Cook W. L. Schmidt R. R. Temps R. A. Laura M. F. Haughey	J. R. Johnson
8/25 - 11/3	88.18	W. Cook W. Schmidt R. Temps M. Banerjee A. Howe	J. R. Johnson
10/4 - 11/17	88.19	W. A. Cook W. L. Schmidt R. R. Temps R. A. Plasse M. Banerjee R. A. Laura H. I. Gregg A. G. Krasopoulos	J. R. Johnson
11/18 - 1/6/89	88.20	W. A. Cook R. R. Temps R. A. Laura R. S. Barkley J. E. Carrasco J. R. Johnson	J. R. Johnson
12/3 - 12/21	88.21	W. A. Cook R. A. Laura	J. R. Johnson
6/13 - 6/17	88.22	H. Gregg W. Cook A. Krasopoulos W. Schmidt	P. Eapen
6/20 - 6/24	88.23	R. Evans W. Hansen G. Lapinsky W. Schmidt C. Sisco A. Sutthoff D. Florek	R. Gallo
8/1 - 8/4	88.24	B. Fox G. Martin K. McBridge G. Arthur	W. Lazarus

NRC Site Inspections

Dates	Insp.#'s	Inspectors	Approved by
8/1 - 8/3	88.25	E. Fox W. Schmidt G. Smith R. Temps	W. Lazarus
7/25 - 7/29	88.26	T. Kosky R. Mathew R. Temps W. Schmidt	C. Anderson
8/1 - 8/5	88.27	R. Loesch A. Weadock W. Schmidt R. Temps	M. Shanbaky
8/22 - 8/26	88.28	J. Jang	W. Pasciak
9/26 - 9/30	88.29	W. Tobin D. Cameron W. Schmidt	R. Keimeg
11/14 - 11/18	88.30	D. L. Caphton	N. J. Blumberg
10/3 - 10/7	88.31	R. Loesch	M. Shanbaky
1'/14 - 11/18	88.32	R. J. Paolino	R. K. Mathew C. J. Anderson
		Also NRC Contractors A. C. Udy - INEL R. VanderBeek - INEL	
11/28 - 12/2	88.33	B. Davidson J. Furia R. Temps R. Laura	W. Pasciak
8/17 - 8/19	88.201	J. Jacobson T. Silko W. Schmidt	E. Brach

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

MAY 29 1981

Docket No. 50-410

RECEIVED
JUN 03 1981
U.S. NUCLEAR REGULATORY COMMISSION

Niagara Mohawk Power Corporation
ATTN: Mr. Gerald K. Rhode
Vice President
System Project Management
c/o Miss Catherine R. Seibert
300 Erie Boulevard, West
Syracuse, New York 13202

Gentlemen:

Subject: Meeting 50-410/81-04

This refers to a meeting held at our request at the Nine Mile Point Nuclear Power Station, Scriba, New York on April 22, 1981 to discuss the findings of the NRC Region I Systematic Assessment of Licensee Performance (SALP) program evaluation board relating to your performance in conducting activities authorized by NRC License No. CPPR-112. This meeting was attended by Messrs. T. E. Lempges, G. K. Rhode and other members of the Niagara Mohawk organization and by myself and members of the Region I staff.

It is our view that this meeting was beneficial and improved mutual understanding of your program and our inspection efforts.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosure will be placed in the NRC's Public Document Room.

No reply to this letter is required; however, should you have any questions concerning this matter, we will be pleased to discuss them with you.

Sincerely,

Boyce H. Grier

Boyce H. Grier
Director

Enclosure: Office of Inspection and Enforcement Inspection Report Number 50-410/81-04 with Appendix and Attachments 1 and 2.

cc w/encl:
Eugene B. Thomas, Jr., Esquire

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ENCLOSURE 1

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

Region I

Report No. 50-410/81-04

Docket No. 50-410

License No. CPPR-112 Priority - Category B

Licensee: Niagara Mohawk Power Corporation

300 Erie Boulevard, West

Syracuse, New York 13202

Facility Name: Nine Mile Point Nuclear Power Station, Unit 2

Inspection at: Scriba, New York

Inspection conducted: April 22, 1981

Inspectors: R. Feil 5-15-81
R. Feil, Reactor Inspector date signed

R. B. Kister 5/15/81
R. B. Kister, Chief, Reactor Projects Section 1C, DR&PI date signed

Approved by: E. J. Brunner 5/22/81
E. J. Brunner, Acting Director, Division of Resident and Project Inspection date signed

Meeting Summary:
Meeting on April 22, 1981 (Report No. 50-410/81-04)
Scope: Special announced management meeting to discuss the results of the NRC board convened to evaluate the Licensee's Performance from February 1, 1981 to January 31, 1981 as part of the NRC's Systematic Assessment of Licensee Performance (SALP) Program. Areas addressed included: Quality Assurance, Substructure and Foundations, Concrete, Liner (Containment and Others), Safety-Related Structures, Piping and Hangers, Safety-Related Components, Electrical Equipment Electrical (Tray and Wire), Instrumentation, Fire Protection, Preservice Inspection, Reporting, Environmental, Training and Management.
Results: A summary of the licensee performance evaluation was presented. No new enforcement actions were identified.

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DETAILS

1. Licensee Attendees

   W. M. Bryant, Manager - Quality Assurance
   T. E. Lempges, Vice President, Nuclear Generation
   S. F. Manno, Project Manager - Unit 2
   D. R. Palmer, Supervisor, Quality Assurance - Operations
   T. J. Perkins, General Superintendent - Nuclear Generation
   G. K. Rhode, Vice President - System Project Management
   T. W. Roman, Station Superintendent - Unit 1
   W. Rumberger, Assistant Project Manager - Operations, Unit 2
   R. W. Smith, Superintendent - Production, Unit 2

2. NRC Attendees

   B. H. Grier, Director, OIE, Region I
   E. J. Brunner, Acting Director, Division of Resident and Project Inspection, Region I
   H. B. Kister, Chief, Reactor Projects Section 1C, DR&PI
   S. D. Hudson, Senior Resident Inspector, Unit 1
   R. A. Fail, Reactor Inspector, Reactor Projects Section 1C, DR&PI

3. Discussion

   A brief summary of the Systemtic Assessment of Licensee Performance (SALP) Program was presented to explain the basis and purpose of the evaluation.

   The NRC Region I evaluation board meeting was discussed, including the assessment period, performance data and the results of the board's evaluation. (See Appendix 1 with Attachment 1 and 2 and Enclosure 2).

   The licensee's overall performance was considered acceptable.

APPENDIX 1

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE

NINE MILE POINT NUCLEAR POWER STATION UNIT 2

REGION I EVALUATION BOARD MEETING

Facility: Nine Mile Point Nuclear Power Station Unit 2

Licensee: Niagara Mohawk Power Corporation

Unit Identification:

Docket No.	License No./Date of Issue	Unit No.
50-410	CPPR-112/June 15, 1972	2

Reactor Information:

NSS: General Electric Co.

MWT: 3323

Assessment Period: February 1, 1980 to January 31, 1981

Evaluation Board Meeting Date: April 1, 1981

Review Board Members:
B. H. Grier, Director, Region I
J. M. Allan, Deputy Director, Region I
E. J. Brunner, Acting Director, Division of Resident and Project Inspection, RI
R. J. Bores, Chief, Independent Measurements and Environmental Protection Section
R. T. Carlson, Director, Enforcement and Investigation Staff, Region I
J. W. Devlin, Chief, Physical Protection Section, Division of Engineering and Technical Inspection, RI

Other NRC Attendees:
H. B. Kister, Chief, Reactor Projects Section 1C, RI
P. J. Polk, Project Manager, Operating Reactors Branch 2, DL, NRR
R. A. Feil, Reactor Inspector, Reactor Projects Section 1C, RI
J. T. Wiggins, Reactor Inspector, Reactor Projects Section 1C, RI

Attachment: Licensee Performance Data

ATTACHMENT 1

NINE MILE POINT NUCLEAR POWER STATION UNIT 2

LICENSEE PERFORMANCE DATA

Assessment Period: February 1, 1980 to January 31, 1981

A. Number and Nature of Noncompliance Items

1. Noncompliance Category

Violations	0
Infractions	1
Deficiencies	0
Severity I	0
Severity II	0
Severity III	0
Severity IV	0
Severity V	0
Severity VI	0
Deviations	0

2. Areas of Noncompliance

Quality Assurance

NINE MILE POINT NUCLEAR POWER STATION UNIT 2

LICENSEE PERFORMANCE DATA

Assessment Period: February 1, 1980 to January 31, 1981

B. Number and Nature of Licensee Event Reports

During the assessment period, 4 construction deficiency reports (CDR's) were submitted by the licensee. Two (2) CDR's were the result of construction/QC errors, one (1) CDR resulted from a vendor/QC error and one (1) CDR resulted from a design/fabrication error. Two (2) CDR's have been resolved but not inspected by IE. The remaining two (2) CDR's are in the process of being resolved by the licensee.

1. Type of Events

   a. Construction/QC Error 2

   b. Vendor/QC Error 1

   c. Design/Fabrication Error 1

2. Causally Linked Events

   None

3. Licensee Event Reports Reviewed (Report Nos.)

   None

C. Escalated Enforcement Actions

Civil Penalties

None

Orders

None

Immediate Action Letters

None

NINE MILE POINT NUCLEAR POWER STATION UNIT 2

LICENSEE PERFORMANCE DATA

Assessment Period: February 1, 1980 to January 31, 1981

D. Management Conferences Held During Past Twelve Months

No enforcement related meetings were held with licensee management during the assessement period.

On May 7, 1980, representatives from Niagara Mohawk Power Corporation (NMPC) and Stone and Webster Engineering Company (S&W) met with Region 1 Construction Branch representatives at King of Prussia, Pennsylvania. The meeting was held at the licensee's request to make a presentation of the biological shield wall weld defect problem.

E. Licensee Activities

Construction activity has been at a reduced level. Estimated completion is 31%. Work force during the assessment period has ranged from 400 - 600 craft, inspection & support personnel. The present schedule for significant events is listed below:

Operating License Application	January 1983
Construction Completion Date	December 1985
Fuel Load Target Date	March 1986

F. Inspection Activities

Eleven inspections were conducted during the assessment period. Areas inspected included: containment structure, quality assurance, reactor vessel installation, reactor vessel internals & biological shield wall. One inspection module is overdue. (Environmental) (Inspection conducted 2/81).

The facility received 380 inspection hours during the assessment period.

G. Investigations

None

ATTACHMENT 2

NINE MILE POINT NULCEAR POWER STATION
UNIT 2

INSPECTION PROGRAM CHANGES

CONSTRUCTION

Inspection Time and/or Scope Changes From Prescribed IE Inspection Program

Functional Area	Increase	No Change	Decrease
1. Quality Assurance		X	
2. Substructure and Foundations		X	
3. Concrete		X	
4. Liner (Containment and Others		X	
5. Safety-Related Structures		X	
6. Piping and Hangers (Reactor Coolant and Others)		X	
7. Safety-Related Components (Vessel, Inernals, and HVAC)		X	
8. Electrical Equipment		X	
9. Electrical (Tray and Wire)		X	
10. Instrumentation		X	
11. Fire Protection		X	
12. Preservice Inspection		X	
13. Reporting		X	
14. Environmental		X	
15. Training		X	
16. Management		X	

Boyce H. Grier
Regional Director

April 1 1981
Date

JUN 18 1982

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. Gerald K. Rhode
Vice President
System Project Management
c/o Miss Catherine R. Seibert
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

Subject: Management Meeting NRC Report No. 50-410/82

This refers to the meeting held at our request at the NRC Region I Office, King of Prussia, Pennsylvania, on June 1, 1982 relating to activities authorized by NRC License No. CPPR-112. This meeting was attended by Mr. G. Rhode and other members of your staff and by myself and other members of the NRC staff. The subjects discussed at this meeting are included in the NRC Management Meeting Report No. 50-410/82-06, which is enclosed with this letter.

It is our view that the meeting was beneficial and improved your understanding of our concerns. Further, it is evident that additional effort on your part is needed to accelerate your review of the matters discussed to, a) fully identify the root cause and b) develop the necessary correction actions. As we discussed, it is our intent to meet with you again within 30 days to discuss your corrective actions.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure will be placed in the NRC Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1). The telephone notification of your intent to request withholding, or any request for an extension of the 10 day period which you believe necessary, should be made to the Supervisor, Files, Main and Records, USNRC Region I, at (215) 337-5223.

Sincerely,

Original Signed By:
R. W. Starostecki, Director,
Division of Project and Resident
Programs

OFFICIAL RECORD COPY

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Q PDR

Enclosure: Management Meeting NRC Report Number 50-410/82-06

cc w/encl:
Leonard M. Trosten, Esquire
Carl D. Hobelman, Esquire
NMPC QA
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
State of New York
bcc w/encl:
Region I Docket Room (with concurrences)
Chief, Operational Support Section (w/o encl)

RI:DPRP	RI:DPRP	RI:DPRP	RI:DPRP
Schulz/pja	Kister	Brunner	Starostecki
6/11/82	6/15/82	6/15	6/15/82

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U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-410/82-06

Docket No. 50-410

License No. CPPR-112 Priority -- Category A

Licensee: Niagara Mohawk Power Corporation

300 Erie Boulevard West

Syracuse, New York 13202

Facility Name: Nine Mile Point, Unit 2

Meeting at: USNRC, Region I, King of Prussia, Pennsylvania

Meeting conducted: June 1, 1982

Inspectors: for HB Kister
R. D. Schulz, Resident Inspector
6/15/82
date signed

Approved by: HB Kister
H. B. Kister, Chief, Reactor Projects Section 1C
6/15/82
date signed

Meeting Summary
Enforcement Conference on June 1, 1982 (Report No. 50-410/82-06)

Summary: Special enforcement conference convened by NRC Region I management to discuss NRC concerns regarding the recent inspection finding related to the incorrect welding procedures authorized for use on Class I piping joints and concerns involving management control of ITT Grinnell by Stone & Webster Engineering Corporation. Senior Niagara Mohawk, Stone & Webster, and NRC Region I management personnel attended the meeting which was held at the Region I office. The meeting was two hours and fifteen minutes in duration.

DETAILS

1. Attendees

Niagara Mohawk Power Corporation

D. P. Dise, Vice President, Quality Assurance
S. F. Manno, Project Manager, Unit 2
G. L. Rhode, Vice President, System Project Management

Stone & Webster Engineering Corporation

R. B. Kelly, Vice President and Manager, Quality Assurance
R. S. Zanetti, Vice President, Assistant Manager, Cherry Hill Operations Center

U. S. Nuclear Regulatory Commission

R. W. Starostecki, Director, Division of Project and Resident Programs
E. J. Brunner, Chief, Reactor Projects Branch No. 1
H. B. Kister, Chief, Reactor Projects Section 1C
R. D. Schulz, Resident Inspector, Nine Mile Point, Unit 2

2. Discussion

Niagara Mohawk began the meeting with a discussion of the upcoming major reorganization in their corporate office. The thrust of the reorganization is intended to place all nuclear activities, both construction and operations, under a newly established position of Senior Vice President, Nuclear Services. Also, the position of Vice President, Quality Assurance will now report directly to the President of the Corporation.

NRC Region I continued by summarizing the reason for the meeting and the intent to provide a forum for discussion of the issues involved.

Niagara Mohawk summarized the event and the actions taken upon discovery of the problem by the resident inspector. Actions taken included issuance of a stop work order to prohibit further welding using incorrect procedures, and directing Stone and Webster to conduct a full audit in this area. Niagara Mohawk further stated that a separate analysis was also performed by ITT Grinnell to determine the acceptability of the affected welds. The conclusion reached by Grinnell was that the welds would probably be acceptable except for verification of actual heat input. Stone and Webster subsequently issued an Engineering and Design Coordination Report to cutout and reweld the affected welds using the correct procedure.

The Stone and Webster Manager of Quality Assurance briefly discussed the results of their "draft" audit findings. The findings appeared to vaguely reinforce NRC Region I findings relating to ITT Grinnell's problems incorporating technical code requirements into Field Planners. The "draft audit" reports apparently found that ITT Grinnell personnel were adept at following a formalized procedure, however, their ability to pickup requirements not specifically spelled out in procedures were suspect.

There also seemed to be some disagreement between NRC Region I findings and Stone and Webster audit findings regarding the qualifications of ITT Grinnell personnel who prepare field planners. Upon comparison of the names reviewed, however, the disparity appeared to have resulted from a difference in the personnel selected for audit. NRC Region I emphasized that the group of twelve people selected were individuals who were involved in preparation of Field Planners. As a result of further questioning it was determined that Niagara Mohawk had not yet had the opportunity to review in detail the Stone and Webster audit findings.

3. Results

After further discussion NRC Region I concluded that neither Niagara Mohawk nor Stone and Webster had fully addressed the relevant issues regarding the root cause of the problems that permitted Field Planners with unqualified welding procedures listed to pass through three levels of review without correction. The relevent issues are considered to be the following.

(1) Inexperience of ITT Grinnell personnel compounded by a lack of ITT Grinnell management involvement commensurate with this inexperience.

(2) Ineffective control by Stone and Webster site management of ITT Grinnell. The resident inspector perceived a significant lack of communication between Stone and Webster site management and ITT Grinnell site management which has resulted in inadequate control over areas requiring interface.

NRC Region I requested that Niagara Mohawk accelerate the issuance of the Stone & Webster audit findings, and vigorously pursue the cause of the event and associated problems. It was further suggested that Niagara Mohawk consider performing their own audits of this area to assure that the problems are fully identified and dealt with. It was further requested that Niagara Mohawk be prepared to reconverse this meeting within 30 days to discuss their findings and corrective actions.

APR 1 1986

Docket No. 50-410

Niagara Mohawk Power Corporation
ATTN: Mr. B. G. Hooten
Executive Director
Nuclear Operations
c/o Miss Catherine R. Seibert
300 Erie Boulevard, West
Syracuse, New York 13202

Gentlemen:

Subject: EXAMINATION REPORT No. 50-410/85-41(OL)

This transmits the Examination Report of Operator Licensing Examinations conducted by USNRC Region I at the Nine Mile Point 2 Facility the weeks of December 9 and 16, 1985. At the exit interview held on December 20, 1985, the preliminary results of these examinations were discussed.

The examiners identified one problem during this examination period that warrants further attention. The NMP training material contained a higher than normal number of inaccuracies which resulted in much confusion for both the candidates and the examiners. These inaccuracies were presented to the NMP training department during both the post-examination review of the written examination and while conducting the operating (simulator) examinations. The NMP Training department must place increased emphasis on correcting the information presented in the lesson plans to ensure consistency with approved operating procedures. If not corrected, this problem will adversely affect the operator licensing examination process in the future.

Details concerning this problem are contained in the body of this report.

No reply to this letter is required. Your cooperation in this matter is appreciated.

Sincerely,

Original Signed By:

Samuel J. Collins, Chief
Projects Branch No. 2
Division of Reactor Projects

A-4

Enclosure:
Examination Report No. 50-410/85-41(OL) w/attachments 1, 2, and 3

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cc w/enclosure and Attachments 1, 2 and 3:
R. B. Abbott, Station Superintendent
R. Zollitsch, Plant Training Manager
Senior Resident Inspector
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
State of New York

cc w/enclosure without Attachments 1, 2 and 3
Connor & Wetterhahn
John W. Keib, Esquire
D. Quanime, NMP-2 Project Director
C. Beckham, NMPC QA Manager
Department of Public Service, State of New York

bcc w/o attachments to enclosure:
DRP Section Chief
Examiner
Chief, OLB, DHFS, NRR
File 120
Region I Docket Room (with concurrences)
Master Exam File
W. Cliff, PNL
B. Hajek, NRC Consultant

RI:DRP	RI:DRP	RI:DRP	RI:DRP	RI:DRP
Kolonauski/ca	Lange	Keller	Linville	Kister
3/21/86	3/21/86	3/21/86	3/21/86	3/26/86

RI:DRP Collins
~~Starostecki~~
3/28/86

OFFICIAL RECORD COPY OL 9MILE2 85-41 - 0001.0.1
03/21/86

EXAMINATION REPORT

Examination Report No. 85-41 OL

Facility Docket No: 50-410

Licensee: Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Facility: Nine Mile Point 2

Examination Dates: December 10-19, 1986

Chief Examiner:	David Lange, Lead BWR Examiner	3/24/86 date
Reviewed by:	Robert Keller, Chief Projects Section 1C	3/25/86 date
Approved by:	Harry Kister, Chief Projects Branch No. 1	3/26/86 date

Summary: This examination report contains the results of the Operator Licensing examinations given at the Nine Mile Point 2 Nuclear Station the weeks of December 9 and 16, 1985. Twelve (12) Senior Reactor Operator candidates and twenty (20) Reactor Operator candidates were examined. All RO candidates passed the written and oral examinations; two (2) failed the simulator examination. Of the SRO candidates, two (2) failed the written examination, one (1) failed the simulator examination, and one (1) failed both the written and oral examinations.

EXAMINATION REPORT

Examination Report No. 85-41 OL

Facility Docket No: 50-410

Licensee: Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Facility: Nine Mile Point 2

Examination Dates: December 10-19, 1986

Chief Examiner:	David Lange, Lead BWR Examiner	date
Reviewed by:	Robert Keller, Chief Projects Section 1C	date
Approved by:	Harry Kister, Chief Projects Branch No. 1	date

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REPORT DETAILS

TYPE OF EXAMS: Initial X Replacement ___ Requalification ___.

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written Exam	20/0	7/3
Oral Exam	12/0	11/1
Simulator Exam	10/2	11/1
Overall	18/2	8/4

1. Chief Examiners at Site: David Lange, NRC
   Lynn Kolonauski, NRC

2. Other Examiners:

   Frank Crescenzo, NRC
   Allen Howe, NRC
   Brian Hajek, NRC Consultant
   Gary Sly, PNL
   William Cliff, PNL
   Lee Miller, NRC

1. Summary of generic strengths or deficiencies noted on oral exams:

   Most candidates were well aware of the differences between the simulator and the plant.

   A few of the simulator groups were deficient in communication skills and procedure usage.

2. Summary of generic strengths or deficiencies noted from grading of written exams:

   No generic strengths or deficiencies were noted on the RO exam.

   An overall weakness was noted in Section 5 of the SRO exam; more specific weaknesses included an unfamiliarity with the Safety Parameter Display System (SPDS) and procedural cautions for the Reactor Recirculation System.

3. Comments on availability of, and candidate familiarization with plant reference material in the control room:

   Most candidates were adequately familiar with plant procedures but several of the simulator groups used only a limited number of procedures.

   The RO candidates were weak in locating specific piping and instrumentation diagrams as requested by the examiners. The SRO candidates, however, were very successful in locating and using the P&IDs.

4. Personnel Present at Exit Interview:

   NRC Personnel

   David Lange, BWR Chief Examiner, Region I
   Allen Howe, Reactor Engineer Examiner, Region I
   Frank Crescenzo, Reactor Engineer Examiner, Region I
   Steven Hudson, Senior Resident Inspector
   Lee Miller, Operator Licensing Branch, HDQ.

   Facility Personnel

   R. T. Seifried, Nuclear Training Assistant Superintendent
   M. D. Jones, NMP 2 Operations Superintendent
   G. L. Weimer, Associate Generation Specialist, Nuclear
   K. F. Zollitsch, Nuclear Training Superintendent
   T. J. Perkins, Nuclear Generation Superintendent

5. Summary of NRC Comments made at exit interview:

The examiners noticed that plant accessibility requirements changed daily; this example of inconsist nt access control may be indicative of a plant security problem.

Several problems caused delays during the written exam:

1. Numerous incorrect and confusing Tech Spec action statements caused approximately a one-half hour delay when answering and clarifying two questions on Section 8 of the SRO exam.

2. The training material (mainly the NMP Lesson Plans and the Q/A Bank) sent to the examiners for preparing the written exam contained many inaccuracies. As a result, the questions prepared from this material were confusing and required a great deal of clarification. Many of the candidates even asked which answer we were looking for - "the one in the lesson plan or the one in the procedure?"

   During the written exam review, the training department asked the exam author to accept both the "right" and the "wrong" answer, because the wrong answer was identified in the training material and the candidates had been exposed to it. We feel accepting incorrect answers for the sake of the training material is not justified and contrary to the interest of safety.

3. The SRO candidates were told to use the TS handout to answer the questions in Section 8 only. The examiners observed several candidates using the TS handout as a "memory jogger" while working in the other exam sections. This resulted in wasted time while searching through Tech Specs and not using the handout as directed.

4. The Learning Objectives identified in the lesson plans should be revised to better represent the operating procedures and actual job performance. Whenever possible, the written exam questions are referenced to learning objectives. The NMP2 lesson plans, however, did not include an adequate amount of learning objectives suitable for this use.

5. A number of errors were identified in the surveillance testing procedures used during the simulator exams; the NMP training department was notified about the errors.

   The examiners thanked the NMP training department for their cooperation during the exam period. The room provided for the examiners next to the simulator was very useful and appreciated.

6. Summary of facility comments and commitments made at exit interview:

   The licensee agreed to send a copy of the Refueling On-the-Job training schedule for the operators to Region I.

7. Changes made to written exam during examination review:

   The following attachment addresses the NRC resolutions of Niagara Mohawk comments on the NMP 2 examinations given on December 10, 1985.

1. Written Examination and Answer Key (RO)
2. Written Examination and Answer Key (SRO)
3. NRC Resolution of Niagara Mohawk Comments

U. S. NUCLEAR REGULATORY COMMISSION
REACTOR OPERATOR LICENSE EXAMINATION

FACILITY	NINE MILE POINT 2
REACTOR TYPE	BWR-GE5
DATE ADMINISTERED	85/12/10
EXAMINER	C. A. SLY
APPLICANT	MASTER KEY

INSTRUCTIONS TO APPLICANT

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE		CATEGORY
25.00	25.00			1.	PRINCIPLES OF NUCLEAR POWER PLANT OPERATION, THERMODYNAMICS, HEAT TRANSFER AND FLUID FLOW
25.00	25.00			2.	PLANT DESIGN INCLUDING SAFETY AND EMERGENCY SYSTEMS
25.00	25.00			3.	INSTRUMENTS AND CONTROLS
25.00	25.00			4.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
100.00	100.00				TOTALS

FINAL GRADE ________________%

All work done on this examination is my own. I have neither given nor received aid.

____
APPLICANT'S SIGNATURE

QUESTION 1.01 (2.00)

The reactor is operating at 75% power. Recirculation flow is subsequently increased to provide 100% power. Briefly EXPLAIN the reactivity transient caused by the flow/power increase with emphasis on the following: (Your answer should include the initial effect, what happens during the power change, and the final steady state.)

a. core vo[illegible]ent (1.0)

b. core reactivity (1.0)

QUESTION 1.02 (1.00)

Concerning control rod worths during a reactor startup from 100% PEAK XENON versus a startup under XENON-FREE conditions, WHICH statement is correct? (1.0)

a. PERIPHERAL control rod worth will be LOWER during the PEAK XENON startup than during the XENON-FREE startup.

b. CENTRAL control rod worth will be HIGHER during the PEAK XENON startup than during the XENON-FREE startup.

c. BOTH control rod worths will be the SAME regardless of core Xenon conditions.

d. PERIPHERAL control rod worth will be higher duri[illegible] the PEAK XENON startup than during the XENON-FREE startup.

QUESTION 1.03 (1.50)

The Reactor has been scrammed following 100 days of full-power operation. STATE whether the following statements concerning fission poisons are TRUE or FALSE.

a. A 25% power reduction from 100% power would have a LARGER Xenon peak than a 25% power reduction from 50% power. (0.5)

b. The Equilibrium Concentration of Samarium IS DEPENDENT on flux level (i.e., stable 100% power or stable 50% power). (0.5)

c. Upon restarting the reactor following a 6-month outage, the Samarium Concentration will DECREASE to its 100% full power concentration. (0.5)

QUESTION 1.04 (1.50)

During a routine startup, control rods are withdrawn, adding a specific amount of reactivity. Consider two (2) cases: 1) that the reactor was slightly subcritical (Keff = 0.995), and 2) that the reactor was greatly subcritical (Keff = 0.95). CHOOSE the word or words that best complete the sentence.

a. The change in the count rate in the slightly subcritical reactor would be (GREATER THAN, LESS THAN, EQUAL TO) the change in the count rate of the greatly subcritical reactor. (0.5)

-Rise-

b. The rise in the count rate in the slightly subcritical reactor would be (FASTER THAN, SLOWER THAN, THE SAME AS) the rise in count rate of the greatly subcritical reactor. (0.5)

c. The time required to reach the equilibrium count rate in the slightly subcritical reactor would be (SHORTER, LONGER, THE SAME AS) in the greatly subcritical reactor. (0.5)

QUESTION 1.05 (2.50)

Nine Mile Pt.-2 Reactor has just experienced a LOCA. An operator wishes to use nuclear instrumentation to determine water level within the core. Your answer should include WHAT nuclear instrumentation would be used, HOW you would use this nuclear instrumentation, WHAT indications you would see and WHY! (2.5)

QUESTION 1.06 (2.50)

During your shift an SRV inadvertently opens from 100% power 1000 psia. By using the Mollier Diagram or Steam Tables,

a. WHAT is the tailpipe temperature assuming atmospheric pressure in the suppression pool? (0.5)

b. If the suppression pool pressure were to increase, WHAT would the tailpipe temperature do (INCREASE, DECREASE, or STAY THE SAME)? (0.5)

c. If the reactor is then depressurized, WILL the tailpipe temperature initially (INCREASE, DECREASE, or STAY THE SAME)? (0.5)

d. At WHAT R_x pressure would the tailpipe temperature be at its maximum value and WHAT temperature is it? (0.5)

e. At WHAT R_x pressure would the tailpipe temperature be at its minimum value? INCLUDE value and assume a saturated system. (0.5)

QUESTION 1.07 (2.50)

STATE, for the following conditions, whether pump ampere would INCREASE, DECREASE, or REMAIN THE SAME:

a. the pump suction valve is slowly throttled closed (0.5)

b. increase in inlet subcooling (0.5)

c. slow closure in the discharge valve of the pump (0.5)

d. rotor lock-up (0.5)

e. rotor failure (break) (0.5)

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.08 (1.50)

Increasing recirculation pump speed will cause WHAT change (INCREASE, DECREASE, or REMAIN THE SAME) in each of the following parameters?

a. actual bundle power (0.5)

b. critical power (not ratio or margin) (0.5)

c. critical power ratio (0.5)

QUESTION 1.09 (2.00)

A "central" and "peripheral" bundle have been inadvertently placed in each others' location. WILL the misplaced bundles power and flow be (HIGHER THAN, LESS THAN, or THE SAME AS) the same type of bundle in the same area of the core?

a. Central bundle in peripheral location: (1.0)

b. Peripheral bundle in central location: (1.0)

<orificing> fixed

QUESTION 1.10 (2.00)

MATCH the most correct "parameter" listed below to the corresponding "fuel integrity item". (2.0)

Parameter

1. LHGR
2. Bulk boiling
3. Total peaking factor (TPF)
4. Onset of transition boiling (OTB)
5. Critical quality
6. CPR
7. APLHGR
8. Boiling length

Fuel Integrity Item

a. Specified to protect against boiling transition.

b. Specified to limit plastic strain and deformation of cladding to less than 1%

c. Specified to limit peak fuel cladding temperature during a LOCA to less than 2200 deg F.

d. Specified as the point/time when the liquid film along the rod's surface is evaporated and cladding temperature starts to rise rapidly.

QUESTION 1.11 (2.00)

COMPLETE the following: (Blanks A through D MAY have more than one word) (2.0)

Xe-135 has two (2) methods of production. About 95% of the Xe is produced by ______(A)______ and the remaining 5% of Xe is produced by ______(B)______. Xe also has two (2) removal methods, at high power levels ______(C)______ is the major removal method, at low power levels ______(D)______ becomes the predominant removal method.

QUESTION 1.12 (1.00)

Explain HOW and WHY excess reactivity varies with core age. (1.0)

(***** CATEGORY 01 CONTINUED ON NEXT PAGE *****)

QUESTION 1.13 (3.00)

You are currently operating at 100% power BOL when you loose partial feedwater heating.

a. If the same situation were to occur at EOL, WHAT would be the corresponding reactivity changes (MORE NEGATIVE, LESS NEGATIVE, NO CHANGE) to each of the ~~above~~ coefficients (i.e., delta (T)-mod, delta (% voids), delta (T))? (1.5)
fuel

b. If the STA tells you that feedwater temperature decreased by 10 deg F, voids decreased by 2%, and reactivity returns to zero, WHAT would be the corresponding temperature change to the fuel temperature? (Assume no rod movement, recirculation flow changes.) (1.5)

(***** END OF CATEGORY 01 *****)

QUESTION 2.01 (2.50)

Concerning the Standby Gas Treatment System

a. ARRANGE the following components in flowpath order from the reactor (1.0)

1. fan
2. demister
3. electric heater
4. flow element (train)
5. radiation element

b. STATE whether the following signals would (INITIATE, ISOLATE or NOT CHANGE) the SBGT system. (1.5)

1. The receipt of a high temperature alarm in Train "A". Assume Train "A" running and Train "B" had been manually stopped. (Answer for each train.)

2. High radiation alarm at the front face of the turbine.

3. High radiation alarm in the HPCS pump room.

4. Water level equal to 105 inches.

5. Drywell pressure equal to 1.85 psig.

QUESTION 2.02 (2.50)

a. WHAT are the differences in modes of operation for the RHS Loops A and B? (0.5)

b. WHAT is the reason for the interlock between the: (1.0)

1. shutdown cooling suction valve and the test return valve?

2. pressure control valve bypass valve (MOV-23A) and Rx pressure?

c. If a LPCI auto initiation function (high drywell) was overridden to realign the system in shutdown cooling mode and another LPCI signal (triple low level) was to come in, WOULD the RHS Loop realign from the shutdown cooling mode to the LPCI mode? EXPLAIN. (1.0)

QUESTION 2.03 (2.50)

a. The LPCS system design has features to protect piping from overpressure. To manually open the LPCS inboard and outboard isolation valves, the correct sequence is (CHOOSE one) (Assume CS pumps are OFF initially and both valves are closed.) (1.0)

1. valve differential pressure (~~900~~ psig, outboard opens then inboard opens old 1

2. valve differential pressure (700 psig, inboard opens then outboard opens

3. valve differential pressure) 700 psig, outboard opens then inboard opens

4. valve differential pressure) 700 psig, inboard opens then outboard opens

b. DESCRIBE the operation of the core spray sparger break detection system. INCLUDE in your answer WHERE pressure is physically sensed, and WHAT delta pressures are sensed. (1.5)

QUESTION 2.04 (1.00)

The Reactor Recirculation Pump seal cartridge assemblies consist of two (2) sets of sealing surfaces and breakdown bushing assemblies. Failure of the No. 2 seal assembly at rated conditions would result in... (CHOOSE one.) (1.0)

a. an increase in No. 2 seal cavity pressure from approximately 500 psig to approximately 1000 psig

b. a decrease in No. 2 seal cavity pressure from approximately 500 psig to approximately 0 psig

c. an increase in No. 1 seal cavity pressure from approximately 500 psig to approximately 1000 psig

d. a decrease in No. 1 seal cavity pressure from approximately 500 psig to approximately 0 psig

QUESTION 2.05 (2.00)

There are two (2) Check Valves located in the discharge line immediately upstream and downstream of the RCIC primary containment line penetration. STATE the two (2) purposes of these Check Valve. (2.0)

QUESTION 2.06 (2.00)

ANSWER TRUE or FALSE for the following:

a. The CRD Drive Water Header pressure is normally maintained at 260 psig above reactor pressure. (0.5)

b. The standby CRD pump auto starts when the running pump trips. If operating pump trips will other one pump auto starts? (0.5)

c. CRDM Accumulators are charged with air from the service and instrument air system. (0.5)

d. Speed Control of the CRDM is accomplished by throttling valves in the hydraulic control units. (0.5)

Question completed hold on it [illegible]

QUESTION 2.07 (2.50)

STATE the following operating temperatures for the Reactor Water Cleanup System:

a. RWCU pump suction temperature (0.5)

b. NRHX outlet temperature (0.5)

c. Filter-demineralizer high temperature alarm (0.5)

d. Filter-demineralizer inlet system isolation temperature (0.5)

e. Return to feedwater temperature (0.5)

QUESTION 2.08 (2.50)

a. WHAT conditions will cause the Div. III (CSH) diesel generator to shutdown during a LOCA condition? (3 required) (1.5)

b. Besides the fuel oil storage and transfer system, WHAT are the other five (5) auxiliary systems necessary for reliable and safe operation? (1.0)

QUESTION 2.09 (1.50)

Concerning combustible gas production following a Loss of Cooling Accident (LOCA)

a. STATE two (2) sources of hydrogen production (1.0)

b. STATE the single source of oxygen production (0.5)

QUESTION 2.10 (2.00)

LIST the four (4) signals which will cause an automatic Recirculation pump downshift from fast to slow speed (2.0)

QUESTION 2.11 (2.00)

Concerning the four (4) vacuum relief lines between the drywell and the suppression chamber

a. In WHICH direction is the flow designed to go? (0.5)

b. WHAT condition(s) do the vacuum relief lines limit or protect against? (1.5)

QUESTION 2.12 (2.00)

LIST the four (4), non-electrical, trips associated with a reactor feed pump (2.0)

(***** END OF CATEGORY 02 *****)

QUESTION 3.01 (3.00)

WHAT are the four (4) anticipatory scrams, HOW is each sensed, and HOW is each one bypassed? (3.0)

QUESTION 3.02 (2.00)

The reactor is at 100% power with the generator synced to the grid. Electrohydraulic Control (EHC) load set is 105%. By using the attached EHC diagram, EXPLAIN WHAT would happen (control valve, bypass valve) in the following circumstances:

a. load limit potentiometer reduced to 95%. (0.5)

b. maximum combined flow limit potentiometer reduced to 95%. (0.5)

c. "A" pressure regulatory X MTR (~~setpoint~~) fails low. (0.5)

d. failure of two (2) bypass valves full open. (0.5)

QUESTION 3.03 (2.00)

ANSWER the following questions based upon the situation described below.

The RRCS is fully operational. The RRCS receives a reactor water low level (105 inches) signal in both complementary logics of an RFCS channel and remains in for 120 seconds. It takes 100 seconds from the initial reactor water low level signal before the APRM level is downscale.

a. WHICH of the four (4) logics integrated into RRCS are actuated at T = 0 seconds? (0.5)

b. WHICH logics are actuated at T = 25 seconds? (0.5)

c. WHICH logics are actuated at T = 98 seconds? (0.5)

d. HOW LONG from T = 0 seconds is it before the RRCS can be reset? (0.5)

QUESTION 3.04 (2.50)

An automatic RCIC initiation has occurred. Subsequently, RCIC injection was automatically terminated due to high reactor water level.

a. WHAT component in the RCIC system functioned to terminate the injection? (0.5)

b. Assuming no operator action, HOW will RCIC respond to a subsequent decreasing water level? (below high water isolation setpoint) (0.5)

c. If an RCIC "Turbine Test" had been in progress when the initial automatic initiation signal had been received, HOW would the system have responded? (0.5)

d. If, following the initiation, the RCIC turbine had tripped on overspeed, COULD it be reset from the Control Room? (0.5)

e. If the RCIC system were lined-up in standby, WHAT would be the functional result of depressing and releasing the manual isolation button. (0.5)

QUESTION 3.05 (2.00)

For each of the following situations, STATE whether the ADS valves will OPEN, CLOSE, or REMAIN IN THE SAME position.

	Initial Condition	Action/Event	
a.	ADS logic initiated with all ADS valves open	Turn off all operating ECCS pumps.	(0.5)
b.	All ADS logic signals initiated 105 sec timer timing out	Push Channel A High DW PRESS SEAL-IN, push button then timer times out	(0.5)
c.	ADS valves closed ~~on B Main Steamline~~. All initiation signals are in. 105 timer just timed out	Failure of the N2 supply system downstream of storage tank (TK4)	(0.5)
d.	SRV keylock control switch (PNL601) for ADS valve in off position	All initiation signal come in and timer times out	(0.5)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.06 (3.00)

EXPLAIN WHAT affect the following failures would have on reactor level. WHY? (Assume 3-element control and Channel A controlling.)

a. 'C' steam line flow signal fails low. (0.75)

b. Channel 'A' reactor level detector signal fails low. (0.75)

c. Lose of RFP lube oil to the 'A' pump servo motor. FW. C.V. (0.75)

d. Inadvertent activation of the setpoint setdown circuitry. (0.75)

QUESTION 3.07 (1.00)

Concerning the four (4) rod display:

a. A control rod is selected for motion and a double X (XX) appears in the rod position window of the four (4) display panel. WHAT does this mean? (0.5)

b. WHAT if a rod were selected and a position window on the four (4) rod display panel, NOT corresponding to the selected selected rod, indicated blank? (0.5)

QUESTION 3.08 (3.00)

Concerning the Intermediate Range Monitors (IRM):

a. If an IRM is reading 7 on Range 9 and the operator down-ranges to range 7, WHAT will the channel reading be? (0.5)

b. WHAT would be the corresponding APRM power level and WHAT trips, if any, will occur? (1.0)

c. Briefly DESCRIBE HOW the IRM system discriminates for gamma signals. Include in your answer the difference between this method and that used for the SRMs. (1.5)

QUESTION 3.09 (3.00)

The Instrument and Service Air systems receive air from a common set of three (3) air compressors.

a. The control switches must be in the auto after stop (green flag) position during normal operations. If the standby compressor started, WHAT would be the consequences of matching the flag to the running status? (0.5)

b. If the air header pressure continued to drop after the standby compresoor started, and the Service Air System isolated, WHAT action is required to restore Service Air? (0.5)

c. If Instrument Air were to be completely lost, in WHAT position would each of the following valves fail? (2.0)

1. scram inlet valves

2. reactor water cleanup filter/demin inlet and outlet valves

3. cooling tower level control valve

4. condenser 4-inch make-up valve (LV-103, Normal make-up)

QUESTION 3.10 (2.00)

WHAT five (5) conditions will cause the Loop Flow Controllers to automatically transfer from Automatic to Manual, when operating in Master Manual control? (2.0)

(***** CATEGORY 03 CONTINUED ON NEXT PAGE *****)

QUESTION 3.11 (1.50)

Given the following data for APRM Channel C:

LPRM Level:	A	B	C	D
Number of LPRMs assigned:	6	5	5	5
Number of LPRMs bypassed:	3	4	0	0

a. If APRM Channel C selector switch on the local (back) panel was placed to the COUNT position, WHAT would be the expected meter reading? (SHOW calculations.) (0.5)

b. Based on the above data, is APRM Channel C operable? ANSWER YES or NO and EXPLAIN WHY. (1.0)

(***** END OF CATEGORY 03 *****)

QUESTION 4.01 (2.50)

During a plant startup and heatup, several actions must be taken as a function of RPV pressure. For EACH of the following actions, GIVE the approximate pressures by which, or above which, the action must be taken accordinn to N2-IOP-101A, Plant Startup.

a. The ADS must be verified operable prior to reactor pressure exceeding ______________ (0.5)

b. Condenser vacuum must be established prior to opening a bypass valve with the vacuum being maintained by the SJAEs. The EHC will open a bypass valve at approximately ________ (0.5)

c. Start a motor driven feedpump when reactor pressure reaches about ______________ (0.5)

d. Transfer the Mode switch to Run after (among verification of other parameters) the steamline pressure has been verified to be greater than ______________ (0.5)

e. RCIC must be determined operable prior to exceeding a reactor pressure of ______________ (0.5)

QUESTION 4.02 (3.00)

ANSWER the following questions concerning the main generator and load changes. USE the attached Power Factor Chart.

a. WHAT would be the operating load (MWe, KVA) limit with a lagging power factor 0.9 and H2 pressure at 30 psig? (0.5)

b. You are operating at a 0.95 lagging power factor with 75 psig H2 and the load dispatcher orders you to drop your power factor to a 0.9 lagging power factor but maintain maximum MWe output. In general, HOW would you change your operating condition? Include in your answer initial conditions (MWe,KVA), a brief discussion of the power change, and the final conditions (MWe, KVA) (2.5)

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.03 (3.00)

According to the start-up procedure:

a. HOW is the SRM/IRM 1/2 decade overlap supposed to be verified? (1.0)

b. If reactor power is 13% and the mode switch is in start-up, SHOULD the reactor have scrammed? WHY? (0.5)

c. HOW is the reactor determined critical (3 conditions)? (1.5)

QUESTION 4.04 (1.00)

During the "steam condensing mode" of RHS, EXPLAIN HOW reactor cooldown rate is controlled. (1.0)

QUESTION 4.05 (1.00)

WHAT reactor conditions and characteristics (four (4) required) influence the point of criticality and the rate at which it is approached during a reactor startup? (1.0)

QUESTION 4.06 (1.50)

A precaution in OP-92, Neutron Monitoring, states that "BWR cores typically operate with neutron flux noise. Care should be taken when operating in this area."

a. WHAT problem can this "noise" create? (0.5)

b. In WHAT specific operating condition is this applicable? (0.5)

c. WHAT actions are required if this condition exists? (0.5)

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

QUESTION 4.07 (2.00)

For the CRD System:

a. PROVIDE the four (4) indications of a sucessful coupling check. (1.0)

b. WHAT immediate operator actions are required on loss of all CRD flow? (1.0)

QUESTION 4.08 (2.00)

ANSWER TRUE or FALSE to the following questions on the Rod Worth Minimizer (RWM) System.

a. If an insert block is present, then three (3) control rod insert errors HAVE occurred and ALL three (3) rods are positioned two (2) even notches past their pull sheet minimum limits. (0.5)

b. When changing Rx power into the RWM operable range, the Rod Group Window WILL display the highest group which has less than three (3) insert errors and at least one (1) rod withdrawn past its minimum limit. (0.5)

c. The select error lamp WILL illuminate whenever the selected rod is not responsible for the current rod block. (0.5)

d. If Rx power is changed such that the RWM system becomes operational with greater than the maximum amount of insert and withdrawal errors present, NO NORMAL rod movement is possible, unless the group contains a control rod causing an insert/withdrawal error. (0.5)

QUESTION 4.09 (3.00)

ANSWER the following questions concerning radiation and radiological control. For a 20-year-old employee with an accumulated occupational dose of 8 rem.

a. WHAT would be the employees maximun federal limit for the quarter? (1.0)

b. COULD this employee be eligible for a life saving action and not violate any federal limits. EXPLAIN (1.0)

c. If the above individual were assigned to assist in the charging of the CRD accumulator (predicted to take 3-hrs) WOULD he/she violate any administrative limits? Radiation Protection stated that a 25 mrem/hr dose exists in the area. (Answer YES or NO, and PROVIDE limit.) (1.0)

QUESTION 4.10 (2.50)

According to Procedure N2-EOP-RL,

a. WHAT precautions must be taken PRIOR TO placing an ECCS system in manual? (1.5)

b. WHAT precautions must be taken WHILE an ECCS system is in manual? (1.0)

QUESTION 4.11 (2.50)

You have been operating at 60% power when one (1) recirculation loop trips. You have been requested to restart the idle loop.

a. According to the Recirculation Procedure, WHAT are the thermal limits that apply to the restart of an idle loop? (1.5)

b. If the idle loop cannot be restarted, COULD you continue to operate with only one (1) recirculation loop for an extended period of time, (i.e., greater than 8 hours)? EXPLAIN (1.0)

(***** CATEGORY 04 CONTINUED ON NEXT PAGE *****)

STION 4.12 (1.00)

LIST the Entry Conditions for Reactor Pressure Vessel (RPV) Water Level Control. (1.0)

(***** END OF CATEGORY 04 *****)
(************* END OF EXAMINATION ***************)

EQUATION SHEET

Where $\dot{m}_1 = \dot{m}_2$

$(\text{density})_1(\text{velocity})_1(\text{area})_1 = (\text{density})_2(\text{velocity})_2(\text{area})_2$

$KE = \frac{mv^2}{2}$ $PE = mgh$ $PE_1+KE_1+P_1V_1 = PE_2+KE_2+P_2V_2$ where V = specific volume
P = Pressure

$Q = \dot{m}c_p(T_{out}-T_{in})$ $Q = UA\,(T_{ave}-T_{stm})$ $Q = \dot{m}(h_1-h_2)$

$P = P_0 10^{(SUR)(t)}$ $P = P_0 e^{t/T}$ $SUR = \frac{26.06}{T}$ $T = \frac{(B-p)t}{p}$

$\text{delta } K = (K_{eff}-1)$ $CR_1(1-K_{eff1}) = CR_2(1-K_{eff2})$ $CR = S/(1-K_{eff})$

$M = \frac{(1-K_{eff1})}{(1-K_{eff2})}$ $SDM = \frac{(1-K_{eff}) \times 100\%}{K_{eff}}$

$\text{decay constant} = \frac{\ln(2)}{t_{1/2}} = \frac{0.693}{t_{1/2}}$ $A_1 = A_0 e^{-(\text{decay constant})x(t)}$

Water Parameters

1 gallon = 8.345 lbs
1 gallon = 3.78 liters

1 ft^3 = 7.48 gallons

Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3

Heat of Vaporization = 970 Btu/lbm
Heat of Fusion = 144 Btu/lbm
1 Atm = 14.7 psia = 29.9 in Hg

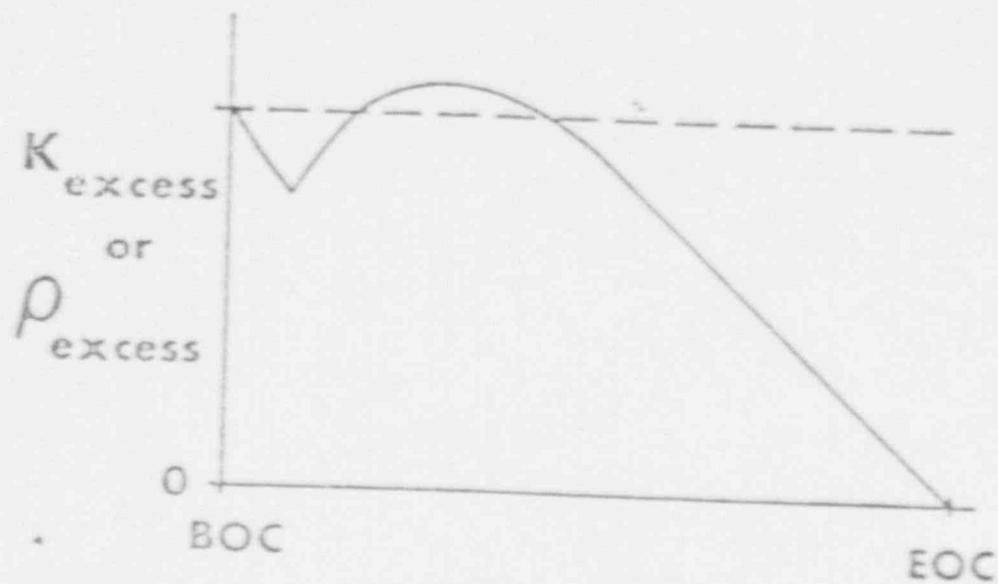
Miscellaneous Conversions

1 Curie = 3.7×10^{10} dps
1 kg = 2.21 lbs

1 hp = 2.54×10^3 Btu/hr

1 MW = 3.41×10^6 Btu/hr
1 Btu = 778 ft-lbf

Degrees F = (1.8 x Degrees C) + 32
1 inch = 2.54 centimeters
$g = 32.174$ ft-lbm/lbf-sec^2



K excess
or
ρ excess
0
BOC
EOC

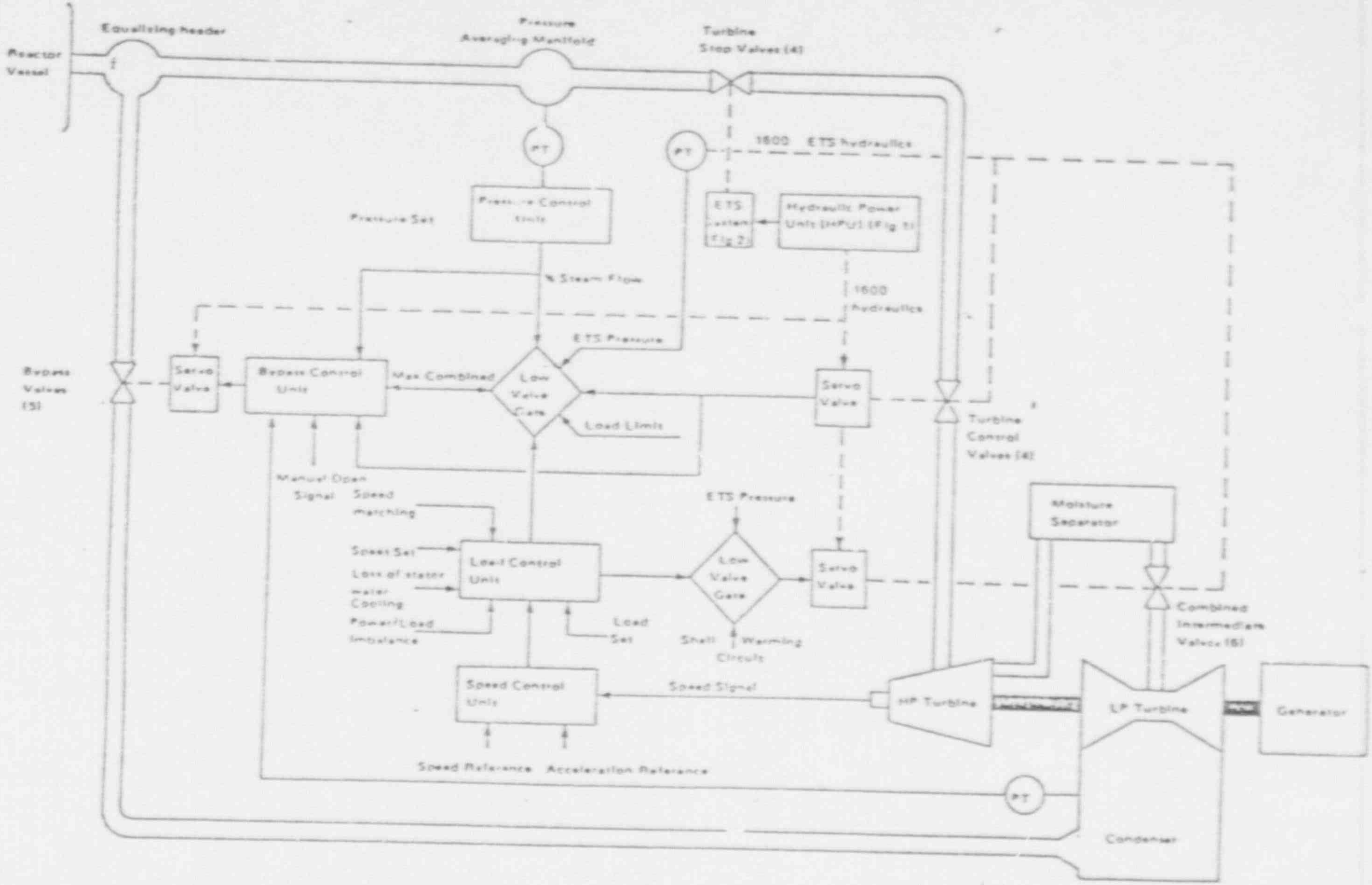


Figure 3	Rev. 0

Title:

TURBINE CONTROL UNIT

2. NMP-2 Exam Bank Question, p. 33, Cat. 1/5

ANSWER 1.07 (2.50)

a. decrease

b. increase

c. decrease

d. increase

e. decrease

(+0.5 pts for each)

REFERENCE

1. NMP-2 Student Learning Objectives for Fluid Statics, Dynamics, and Delivery No. 7, 10, 12, 14, pp. 15 to 17.

ANSWER 1.08 (1.50)

a. actual bundle power increases (+0.5)

b. critical power increases (+0.5)

c. critical power ratio decreases (+0.5)

REFERENCE

1. NMP-2 Student Learning Objective for BWR Thermodynamics and Thermal Hydraulic Limits, No. 7, p. 9

2. General Electric Thermodynamics, Heat Transfer, and Fluid Flow, MTC, March 1983, pp 9-81, 9-86, 9-92.

ANSWERS -- NINE MILE POINT 2 -85/12/10-G A SLY

Since orfices on [illegible] flow would effect flow
∴ Central = lower [illegible] power
Peripheral = [illegible] power

ANSWER 1.09 (2.00)

a. Central bundle in peripheral location

power higher than (+0.5)

flow ~~higher~~ lower than (+0.5) If responds "increase" reverse due to 2 ϕ

b. Peripheral bundle in a central location

power lower than (+0.5) " " "decrease" " "

flow ~~Lower~~ higher than (+0.5)

REFERENCE

1. Thermodynamics Lesson Plan, BWR Thermodynamics and Thermal Hydraulic Limits, p. 13 of 20

2. NMP-2 Examination Bank, Category 1.5, p. 73

ANSWER 1.10 (2.00)

a. 6. CPR - protect against transition boiling

b. 1. LHGR - maintain cladding than 1% plastic strain

c. 7. APLHGR - maintain peak cladding surface temperature to less than 2200 deg F following a LOCA

d. 4. OTB

(+0.5 pts. for each response)

REFERENCE

1. NMP-2 Introduction to Thermodynamics and Thermal Hydraulic Limits, Figure 4, p. 6, Student Learning Objective Nos. 1, 2, 5

ANSWER 1.11 (2.00)

A. I-135 decay (+0.5)

B. Direct fission yield (+0.5)

C. Burnout (+0.5)

D. Xe-135 decay (+0.5)

REFERENCE

1. NMP-2 Operations Technology, Module 1, Part 16, pp. I-16-1 to I-16-3, Student Learning Objective No. 1

ANSWER 1.12 (1.00)

Initially the excess reactivity of the core will decrease due to a buildup of fission product poisons (+0.25). Once fission product poisons reach an equilibrium value, the excess reactivity will increase as burnable poison burnout exceeds fuel burnup (+0.25). This increase continues to a maximum value where fuel burnup begins to exceed poison burnout (+0.25). The value then decreases until refueling due to fuel burnup (+0.25).

REFERENCE

1. NMP-2 Operations Technology, Module 1, Part 7, p. 7-7, Student Learning Objective No. 3. Provide K-excess Graph

2. NMP Examination Bank Question Category 1. 5, p. 39

ANSWER 1.13 (3.00)

a. delta(ro) = alpha(x) [delta(x)]

delta(ro)D = more negative (+0.5)

delta(ro)V = less negative (+0.5)

delta(ro)M = less negative (+0.5)

b. alpha(d)(delta(T)fuel) = alpha(m)(delta(T)mod) + alpha(v)delta(%V) (0.25 pts for equation)

alpha(m) = -1.0 x 10E-5 delta K/K/deg-F (0.25)

alpha(d) = -1.0 x 10E-5 delta K/K/deg-F (0.25)

alpha(v) = -1.0 x 10E-3 delta K/K/%V (0.25)

delta(T)fuel=[(-1 x 10E-4(-10))+(-1 x 10E-3(-2))]/(-1.~~2~~ 1.0 x 10E-5)

delta(T)fuel = -250 deg F or 250 deg F increase in fuel temp (0.5)
-300

REFERENCE

1. NMP-2 Operations Technology Module I, Part 13, pp. 13-5, 13-6, Student Learning Objectives No. 2.c, 3

2. NMP-2 Operations Technology Module I, Part 12, pp. 12-5, 12-7, Figures 12-6, 12-7, Student Learning Objectives No. 2.b, 3.a, 3.b.

ANSWERS -- NINE MILE POINT 2 -85/12/10-C A. SLY

ANSWER 2.01 (2.50)

a. 2, 3, 1, 4, 5
(demister, electric heater, fan, flow element, radiation element) (+0.2 for each)

no change ~~initiate~~ Train "B", no change ~~isolate~~ Train "A" (+0.5) (need high-high
no change (+0.25)
no change (+0.25)
initiate (+0.25)
initiate (+0.25)

REFERENCE
1. NMP-2 Operations Technology, Module V, Part 6, SBGT, pp. 2-7, Student Learning Objective Nos. 2, 3, 5, 6.
N2-IOP-6.B, pg. 7.

ANSWER 2.02 (2.50)

Motor operated valves to loop 'B' only
a. RHR 'B' - head spray (+0.25)
- containment flooding (+0.25) (+0.5 TOTAL)
design change - NOT "modes" exactly

b. 1. Prevent inadvertent draining of the vessel (+0.5)

2. Prevent exceeding RHS design pressure (+0.5) (+1.0 TOTAL)

NO - suction valves do not realign (No auto open on Sup Pool Cooling)
c. ~~Yes~~ (+0.25), because the second LPCI initiation signal will realign the system by reopening the LPCI injection valve. (+0.75) (+1.0 TOTAL)

REFERENCE
1. NMP-2 Operations Technology, Module IV, Part 5, RHS, pp. 5, 9, 10, Student Learning Objective Nos. 1, 5, 6.

ANSWER 2.03 (2.50)

a. "1" (+1.0)

b. Differential pressure is sensed between the core spray injection line (+0.25) and the RHS A LPCI injection ~~nozzle~~ line (+0.25). A break in the CS piping outside the shroud (+0.25) but inside the vessel (+0.25) would cause the dp to increase to ~~to~~ the pressure drop across the steam separators (+0.5, +1.5 Total)

or Jet Pump / Core

REFERENCE

1. NMP-2 Operations Technology, Module IV, CSL, pp 7, 8

ANSWER 2.04 (1.00)

b (+1.0)

REFERENCE

1 NMP-2 Operations Technology, Module III, Recirc., p. 3, Fig. 6

ANSWER 2.05 (2.00)

Prevent back flow (+0.5)

Check Valve - Assure a non-isolatable or non-servicable flow path for RCIC ~~(+1.0)~~ (+0.5) and provides for the inside and outside primary containment isolation valve. (+1.0)

did solicit CV vs. any other valve, will not get isolations

REFERENCE

1 NMP-2 Operations Technology, Module IV, RCIC, p. 3, Fig. 1

ANSWER 2.06 (2.00)

a. TRUE (+0.5)

b. TRUE (+0.5)

c. FALSE (+0.5)

d. TRUE (+0.5)

REFERENCE

1 NMP-2 Operations Technology, Control Rod Hydraulics Rev. 1, p. 10 of 14 Student Learning Objective No

ANSWER 2.07 (2.50)

a. 533 deg F +/- 5 deg F — 1000 psia — 545°F ± 5°F steam table

b. ~120 deg F ~~(no tolerance)~~ O.P. 37 says "about 120°F"

(Now changes "e")

c. 130 deg F (no tolerance)

d. 140 deg F (no tolerance)

e. 437 deg F +/- 5 deg F

Gives you about 444°F

NOTE: Since students were taught that there is a 100 deg F difference between (a) and (e) above, accept a 100 deg F difference for full credit for (e)

(+0.5 for each value)

REFERENCE

1. NP-2-OP-37, Reactor Water Cleanup.

2. NMP-2 Exam Bank

ANSWER 2.08 (2.50)

a. 1 Engine (or generator) overspeed (+0.5)
   2 Generator differintial lockout (+0.5)
   3 Manual stop (+0.5)

b. 1 Fuel oil supply
   2 Jacket water cooling / SW
   3 Starting air
   4 Lubrication
   5 Combustion air (+0.2 each, +1.0 Total)

Acc'd to OP 100.1 pg 7:
1. Station Electrical 115 kV switchyard
2. Sby and Emerg AC dist
3. HPCS 125 Vdc dist
4. DG Bldg ventilation
5. Service Water

REFERENCE

1. NMP-2 Operation Technology, CSH Diesel Generator, Rev. 1, p. 3, 13 of 16

taught "Op Techs for Info only - you run the plant w/ OPs"

ANSWERS -- NINE MILE POINT 2 -85/12/10-G A SLY

ANSWER 2.09 (1.50)

a. radiolytic decomposition of water, zirc-water reaction (+1.0)

b. radiolytic decomposition of water (+0.5)

Add:
1. oxidation of zinc w/in contain
2. O_2 of steel from RPV
Ref: from MCD text

REFERENCE
1. N2-IOP-62 Hydrogen Recombiner

ANSWER 2.10 (2.00)

- List 4 of 5 Below -

a. Steam line to pump suction temp difference is < 7 deg

b. Total feed flow < 30%

c. (EOC-RPT) TSV or TCV closure with power > 30% of rated

d. Reactor water < level 3
(+0.5 for each, +2.0 TOTAL)

e. Rx pressure > 1050 psi via ATWS circuitry
Ref: RRCS Chapter

REFERENCE
1. NMP-2 Operations Technology, Recirculation System, Rev 1, pg 12

ANSWER 2.11 (2.00)

a. Suppression pool to drywell (+0.5)

b. limits negative pressure differential (+0.5) to prevent drawing water up the downcomer from the suppression pool to the drywell (+1.0)
or protect the floor

REFERENCE
1. NMP-2 Operations Technology, Primary Containment, Rev 1 pg 6

SRO - see Q6.13!

ANSWER 2.12 (2.00)

1. RPV high level
2. Low suction w/TD
3. low-low suction
4. low lube oil pressure
(+0.5 for each, +2.0 TOTAL)

REFERENCE

1 NMP-2 Operations Technology. Feedwater Sys., p. 6

ANSWERS -- NINE MILE POINT 2 -85/12/10-G A SLY

ANSWER 3.01 (3.00)

a. TCS, TCV, MSIV closure, plus the SDV high water level scram. (+1.0 TOTAL)

b. TSV - valve position (>5% closed)
TCV-EST fluid pressure (<530 psig)
MSIV closure-valve position (>6% closed)
SDV-level switches (or transmitters) (25 gal) = 46.5"

(+1.0 TOTAL)

c. TSV - bypasses when <30% of rated (1st stage shell pressure)
TCV - same as above
MSIV closure - bypasses with the MSS out of run
SDV scram - MSS in S/D or Refuel with the bypass switches in B/P (4 total)

(+1.0 TOTAL) -0.25 for extra

REFERENCE

1. NMP-2 Operations Technology, Reactor Protection System, Rev 1, Table 1

2. NMP-2 Technical Specification Bases, RPS LSS

ANSWER 3.02 (2.00)

a. control valves close 5% (+0.25), open one bypass valve (+0.25) (or similar answer on diagram) (+0.5 Total)

b. control valves close 5% (+0.25), reactor scram probable due to increasing pressure since bypass valves will not be open (+0.25) (+0.5 Total)

c. will develop a pressure error of 800 psid. This will be a demand for maximum opening of all valves. However, due to the action of the maximum combined flow limiter, control valves will go to ~~105%~~ 100%, demand and bypass valves to ~~10%~~ 15% (+0.5)

d. control valves close to 90% (+0.25) to maintain Rx pressure at 920 psig (+0.25) (+0.5 Total)

REFERENCE

1. NMP-2 Operations Technology, EHC, Rev. 1, pp. 2, 5 to 9 of 14, Student Learning Objective Nos. 5, 6, 8, including EHC Figure 3

ANSWER 3.03 (2.00)

a. alternate rod insertion (+0.5)

b. none (+0.5)

c. standby liquid control (+0.5)

d. ~~80 seconds~~ (+0.5)
11.6 min. or 11 min 35 sec. f SLC

REFERENCE

1. NMP-2 Operations Technology, Module VI, Part 8, pp. 2, 4, 7, Student Learning Objective No. 4

called "steam admission valve"

ANSWER 3.04 (2.50)

a. closure of the turbine stop valve (MOV-120) (+0.5)

b. open the turbine stop valve and reinject (+0.5)

c. align in RCIC mode and inject (+0.5)

d. no (locally) -mechanical- (+0.5) - or - yes electrical can

e. no change to system logic (+0.5)

REFERENCE

1. NMP-2 Operations Technology, Module IV, Part 6, RCIC, pp. 4, 9, 10, Student Learning Objective Nos. 1, 3a, 3b, 5, 6

ANSWER 3.05 (2.00)

a. valves close (+0.5)

b. valves open (+0.5) ~~remove [illegible]~~ [illegible] closed if slotted removed.

c. valves open (0.5)

d. valves open (+0.5)

REFERENCE

1. NMP-2 Operations Technology, Module IV, Part 3, ADS, pp. 10, 12, 13, 16, Student Learning Objective Nos. 3, 4, 5, 6, 7a

ANSWER 3.06 (3.00)

a. decrease (+0.25) due to steam/feed mismatch requiring less water (+0.5) (+0.75 Total)

b. increase (+0.25) due to level mismatch requesting more water (+0.5) (+0.75 Total)

c. no change (+0.25), servo would lock up valve as is (+0.5) (+0.75 Total)

d. decrease (+0.25), due to reduction in operator setpoint of one-half input value (+0.5) (+0.75 Total)

REFERENCE

1. NMP-2 Operations Technology, Module IX, Part 6, pp. 4, 5, Student Learning Objective Nos. 6, 7

ANSWER 3.07 (1.00)

a. bad position indication into the RPIS (+0.5)

b. peripheral rod selected (+0.5)

REFERENCE

1. NMP-2 Operation Technology, Module VI, Part 6, RMCS, p. 6

ANSWER 3 08 (3 00)

a off-scale high. (40) (+0 5)

b 2 8% power (93 1 MWt) (-0.1 each)
rod withdrawal block trip
upscale alarm trip
upscale trip (1/2 scram)
(+0 25 for each, +1 0 TOTAL)

is rltv to APRM ⇒ downscale (+.40)
rod block (0.35)

c The IRM system uses the method of Cambelling to elimimate the gamma signal(+0 5) where as the SRM system uses a pulse height discriminator (+0 5) The Cambelling method, roughly, squares the signal an' then chops the gamma out (+0 5)

REFERENCE

1 NMP-2 Operations Technology, Module VI, Part 2, IRM, pp 3, 4, 6, 8, 10, Student Learning Objective No 3

ANSWER 3 09 (3 00)

a The compressor would not shut down and if it did shut down, it would not automatically restart (+0 5)

b The isolation valve (AOV-171) must be locally reopened This is done by placing a local switch to open (It will spring return to normal. (If the air header pressure is greater than 85 psig, the valve will open) (+0 5)

Not req'd.

c 1 open (+0 5)

2 shut (+0 5)

3 shut (+0 5)

4 open (+0 5) (+2 0 TOTAL)

REFERENCE

1 NMP2 IOP-19, pp 7-10

ANSWERS -- NINE MILE POINT 2 -85/12/10-G.A. SLY

ANSWER 3.10 (2.00)

1. Any initiation of high to low recirc pump speed transfer 1.
2. High Drywell pressure (1.69 psig)
3. Loss of feedpump with concurrent vessel low water level
4. Excessive rate of change of the Flux Controller output
5. Deviation of 1% between the Loop Controller input and manual output signal (tracking failure)

(~~+0.2~~ 0.4 for each, +2.0 TOTAL)

Ref: 2.10 for pump trips

REFERENCE

1. NMP-2 Lesson Plan for RRFCS, pp. 23-24 12 of 26.

ANSWER 3.11 (1.50)

a. ~~60%~~ 70% (+0.5), 5% (volts) for each LPRM not bypassed (+0.5)

(+1.0 Total)

b. No (+0.5). There are fewer than two (2) operable inputs on Level B (+0.5) (+1.0 Total)

REFERENCE

1. NMP-2 Operations Technology, Module VI Part 4, APRM, p. 5, Student Learning Objective Nos. 3, 4

ANSWERS -- NINE MILE POINT 2 -85/12/10-C A SLY

ANSWER 4.01 (2.50)

a. 100 psig
b. 150 psig
c. 200 psig below the condensate booster pump discharge pressure
756 d. ~~765~~ psig
e. 150 psig (+0.5 for each, +2.5 TOTAL)

REFERENCE
1. NMP-2, N2-IOP-101a, Sections E. 2.24, 2.26, 2.30, and 3.5

ANSWER 4.02 (3.00)

a. From the Power Factor Chart or Precautions
0.813 MWe (+0.25) and 400 KVAR (+0.25)

b. Initial State
1,280 MWe (+0.25) and 420 KVAR (+0.25)

Reduce generator load by recirc. or control rods to 1.21 MWe (+0.5). Then raise reactive load (VARs) by adjusting the AC voltage regulator (+0.5). To be done in this order as not to exceed operational limits (+0.5)

Final State
1,21k MWe (+0.25) and 600 KVAR (+0.25) Asks "Why" NOT asked for by question

REFERENCE
1. NMP-2 N2-IOP-68, Main Gen., p.5 and Figure 3. Provide Power Factor Chart

ANSWERS -- NINE MILE POINT 2 -85/12/10-G.A. SLY

ANSWER 4.03 (3.00)

a. By visually observing that all IRMs are above downscale before any SRM count rate is above 10E+5 cps with the SRMs fully inserted (+1.0)

b. No (+0.25), setpoint is 15% power (+0.25) (+0.5 TOTAL)

c. 1. Neutron count rate increasing at a logarithmic rate (+0.5)

2. No control movement (+0.5)

3. A stable positive period. (+0.5)
(+1.5 TOTAL)

REFERENCE
1. N2-IOP-101A, Plant Startup, pp. 8, 13

ANSWER 4.04 (1.00)

Reactor cooldown rate is controlled by the RHS heat exchanger level (+0.5). If the level is reduced, more heat exchanger tubes are exposed, and the condensing of reactor steam increases (+0.25). If the level in the Hx is increased, the condensing rate decreases (+0.25) (i.e., cooldown rate decreases) (+1.0 TOTAL)

REFERENCE
1. N2-OP-31, Residual Heat Removal, H.4, Steam Condensing Mode

ANSWER 4.05 (1.00)

Any four (4) of the following (+0.25 each, max +1.0)

1. Xenon concentration

2. Modarator temperature

3. Control Rod position (axial)

4. Order of Rod Withdrawal

~~5. Core Exposure~~ delete from key

REFERENCE

1. N2-OP-101A, Plant Startup, Precautions, pp 2, 3

ANSWER 4.06 (1.50)

a. High neutron flux alarm and/or scram (+0.5)

b. At or near 100% rod line, minimum recirc flow (+0.5)

c. Insert control rods per Reactor Analyst or increase recirculation flow (+0.5)

REFERENCE

1. N2-OP, Neutron Monitoring, Precautions and Off-Normal Procedures

ANSWERS -- NINE MILE POINT 2 -85/12/10-G.A SLY

ANSWER 4.07 (2.00)

a. A coupling check is the application of a continuous withdraw signal with the control rod full out to check coupling mechanisms by observing the following:

1. red rod "full-out" light remains on (+0.25)

2. rod overtravel annunciator does not come in (+0.25)

3. drive water flow decreases to "stall flow" (+0.25)

4. rod remains at position 48. (+0.25)

(+1.0 TOTAL)

b. 1. Reduce Recirc Flow to minimum (+0.25)

2. Scram the Plant (+0.25)

3. Follow Procedures (+0.25)

4. Notify SSS (+0.25)

(+1.0 TOTAL)

REFERENCE

1. N2-IOP-30, CRD, pp. 21, 22, 30.

2. NMP-2 Exam Bank Cat. 4, CRD, No. 9 (Part A)

ANSWER 4.08 (2.00)

a. TRUE (+0.5) → ? False

b. ~~FALSE~~ (+0.5) True

c. TRUE (+0.5)

d. ~~TRUE~~ (+0.5) False

REFERENCE

1. NMP-2, Operation Technology, Module VI, Part 6, RMCS, pp. 9, 11, Student Learning Objective No. 5

2. N2-IOP-95A, pp. 2, 4

ANSWERS -- NINE MILE POINT 2 -85/12/10-G.A. SLY

ANSWER 4.09 (3.00)

a. 2 rem. (+0.5) due to 5(N-18) limit (+0.5)
b. No (+0.5), exceeds 5(N-18) limit (+0.5)-OR- Yes, once in a lifetime
c. No (+0.5), administrative limits state you can receive 100 mrem per week (+0.5)

REFERENCE
1. NMP-2, S-RP-1, Access and Radiological Control, pp. 1, 12, 17
2. NMP-2, EPP-15, Health Physics Procedure, p. 3

ANSWER 4.10 (2.50)

a. Do not secure or place an ECCS in MANUAL mode unless, by at least two indepencent indications (+0.5),

1. misoperation in AUTOMATIC mode is confirmed (+0.5) or

2. adèquate core cooling is assured (+0.5) (+1.5 TOTAL)

b. If an ECCS is placed in MANUAL mode, it will not initiate automatically. Make frequent checks of the initiating or controlling parameter (+0.5). When manual operation is no longer required, restore the system to AUTOMATIC/STANDBY mode if possible (+0.5). (+1.0 TOTAL)

REFERENCE
1. NMP-2, N2-EOP-RL, RPV Water Level Control, p. 3

ANSWERS -- NINE MILE POINT 2 -85/12/10-G A SLY

ANSWER 4.11 (2.50)

a 1. Steam dome space to bottom drain less than or equal to 145 deg F (+0.5)

2. Idle loop to operating loop less than or equal to 50 deg F (+0.5)

3. Idle loop to bottom drain ~~operating loop~~ less than or equal to 50 deg F ~~% of rated flow~~ (+0.5) (+1.5 TOTAL)

b No (+0.5), required to shutdown (+0.25) due to ECCS performance criteria (i.e., flow imbalance, etc.) (+0.25) (+1.0 TOTAL)
ie. single loop analyses haven't been done yet.

REFERENCE

1 Tech Spec 3/4.4.1.4, pp 3/4 4-4 and B.3/4 4-1

ANSWER 4.12 (1.00)

1 RPV water level less than 159.3 in

2 RPV pressure greater than 1037 psig

3 Drywell pressure greater than 1.68 psig

4 A condition that requires an MSIV isolation

5 A condition that requires an Rx scram, AND Rx power is above 4% or cannot be determined

(+0.2 each, max +1.0)

REFERENCE

1 NMP-2, N2-EOP-RL, RPV Water Level Control, p 1, Student Learning Objective No 2

#

QUESTION 7.06 (2.00)

Following a required initiati[illegible] the S[illegible] Liqu[illegible] System you are directed by the level/powe[illegible] trol [illegible]

a. "Lower RPV water level by terminating [illegible] inject[illegible] except from CRD and Boron i[illegible] until either [illegible]" WHAT is the purpose [illegible] water level at this time? (1.0)

b. You are also directed to inject boron [illegible] suppression pool temperature reaching 110 deg F. [illegible] is [illegible] for boron injection prior to reaching this temperat[illegible] limit? (1.0)

QUESTION 7.07 (1.50)

While attempting to line up shutdown cool[illegible] mode [illegible] your reactor operator informs [illegible] that the [illegible] from the recirculation line is [illegible] in the closed [illegible] According to the Alternate Shutdown Cooling Procedu[illegible] (N2-EOP-[illegible]), in general, WHAT would be the alternat[illegible] [illegible] path for perform[illegible] the [illegible] cool[illegible] (1.5)

QUESTION 7.08 (2.50)

[illegible]erning the blowdown and re[illegible]ulation [illegible] shutd[illegible] modes of the Reactor Water Cleanup (RWCU) System

a. The operator is cautioned [illegible] place [illegible] sy[illegible] [illegible] mode prior to st[illegible] the [illegible] the reason for this cauti[illegible] (0.5)

b. When operating in the blo[illegible] mode [illegible] divert all the RWCU flow [illegible] Main Condenser? (1.0)

c. WHEN and WHY would the [illegible] [illegible]ely be used? (1.0)

QUESTION	VALUE	REFERENCE
01.01	2.00	SLY0000004
01.02	1.00	SLY0000005
01.03	1.50	SLY0000006
01.04	1.50	SLY0000007
01.05	2.50	SLY0000009
01.06	2.50	SLY0000010
01.07	2.50	SLY0000011
01.08	1.50	SLY0000012
01.09	2.00	SLY0000013
01.10	2.00	SLY0000014
01.11	2.00	SLY0000001
01.12	1.00	SLY0000002
01.13	3.00	SLY0000003
	25.00	
02.01	2.50	SLY0000054
02.02	2.50	SLY0000055
02.03	2.50	SLY0000057
02.04	1.00	SLY0000058
02.05	2.00	SLY0000059
02.06	2.00	SLY0000061
02.07	2.50	SLY0000062
02.08	2.50	SLY0000063
02.09	1.50	SLY0000064
02.10	2.00	SLY0000113
02.11	2.00	SLY0000114
02.12	2.00	SLY0000115
	25.00	
03.01	3.00	SLY0000039
03.02	2.00	SLY0000040
03.03	2.00	SLY0000041
03.04	2.50	SLY0000042
03.05	2.00	SLY0000043
03.06	3.00	SLY0000044
03.07	1.00	SLY0000045
03.08	3.00	SLY0000047
03.09	3.00	SLY0000048
03.10	2.00	SLY0000049
03.11	1.50	SLY0000050
	25.00	
04.01	2.50	SLY0000056
04.02	3.00	SLY0000065
04.03	3.00	SLY0000066
04.04	1.00	SLY0000067
04.05	1.00	SLY0000069

STION	VALUE	REFERENCE
4.06	1.50	SLY0000070
4.07	2.00	SLY0000071
4.08	2.00	SLY0000072
4.09	3.00	SLY0000073
1.10	2.50	SLY0000101
4.11	2.50	SLY0000102
4.12	1.00	SLY0000104
	25.00	
	100.00	

Attachment 2

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: NINE MILE POINT 2

REACTOR TYPE: BWR-GE5

DATE ADMINISTERED: 85/12/10

EXAMINER: G. A. SLY

APPLICANT: MASTER KEY

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
25.00	25.00	______	______	5. THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
25.00	25.00	______	______	6. PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
25.00	25.00	______	______	7. PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00	______	______	8. ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00	100.00	______	______	TOTALS

FINAL GRADE ______________%

All work done on this examination is my own. I have neither given nor received aid.

____
APPLICANT'S SIGNATURE

QUESTION 5.01 (2.00)

ANSWER if the following Sm-149 statements are TRUE or FALSE? (2.0)

a. It is REMOVED from an operating reactor by burnout and radioactive decay

b. WHEN a reactor is restarted after a temporary shutdown, Sm-149 concentration increases for several days

c. It has LESS effect on reactor operation than Xe-135 due to its smaller fission yield and smaller microscopic neut[illegible] cross section

d. The equilibrium concentration of Sm-149 at 50% FP is abou[illegible] TWO-THIRDS that of the equilibrium concentration [illegible] 100% F[illegible]

QUESTION 5.02 (2.00)

STATE whether the following situations would (INCREASE, DECREA[illegible] or NOT CHANGE) control rod worth

a. Restart 10 hr following a scram from 100% power condition (peripheral rod only) (0.5)

b. Second rod in a rod group following the withdrawal of the first rod in that group (0.5)

c. Change from a cruciform shaped rod to a cylindrical rod of the same volume (0.5)

d. Localized voiding of region not previously voided (0.5)

QUESTION 5.03 (2.00)

The reactor is critical at 10E+6 cps. A stable period of 60 seconds is achieved. If rods are inserted continuously until the period drops to infinity and then the rod insertion is immediately stopped WILL the reactor be (critical, supercritical, or subcritical) in the time following the rod stoppage? EXPLAIN. (2.0)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.04 (2.50)

You are the SRO in charge of the initial fuel loading process. As part of your duties, Operations has asked you to verify the STAs prediction to criticality as fueld is being loaded.

a. From the following information, PREDICT the point of criticality after the 6th fuel bundle. (1.0)

Count Rate, (cps)	1/M Value
S, CRo = 100	1.00
F1, CR1 = 100	1.00
F2, CR2 = 102	0.97
F3, CR3 = 105	0.95
F4, CR4 = 110	0.91
F5, CR5 = 113	0.87
F6, CR6 = 125	0.80
F7, CR7 = 149	0.67
F8, CR8 = 200	0.50
F9, CR9 = 500	0.20

b. HOW MANY fuel bundles may be loaded following the 6th fuel fuel bundle, prior to being required by ANS/ANSI Standards to make another criticality calculation? (ANS/ANSI Standards state that "... the maximum fuel load increment is the greater of one fuel assembly, or one-half the additional bundles which are predicted to produce criticality.") (1.0)

c. DO the initial six (6) fuel bundles of the 1/M plot indicate that fuel is being loaded (IDEALLY, AWAY FROM the detector or TOWARDS the detector)? (0.5)

QUESTION 5.05 (2.50)

Given a large vented tank 30 ft in diameter and 60-ft high with a centrifugal pump taking a suction from its base. The pump is located at a vertical elevation corresponding to the bottom of the tank and it requires 5 ft of net positive suction head (NPSH) to prevent cavitation. The tank is entirely full of water and is maintained at 60 deg F by heaters. The tank is designed such that it could withstand 15 psi differential pressure in either direction. Assume the vent becomes totally clogged while the pump is in operation. ANSWER the following questions.

a. WHAT is the lowest pressure that the tank will drop to as the pump continues to remove water from the tank? (0.5)

b. WILL the pump loose NPSH and begin to cavitate prior to reaching a level of 5 ft in the tank? EXPLAIN. (State any assumptions.) (1.0)

c. COULD the pump continue to pump water at a level below 5 ft without cavitation if the vent were open? EXPLAIN. (Assume no vortexing.) (1.0)

QUESTION 5.06 (2.00)

Given the following two (2) conditions and using the supplied information, DETERMINE which condition is operating MORE CLOSELY to its MCPR limit. (Show all work and state any assumptions.) K-f graph is provided. (2.0)

Condition 1	Condition 2
Rx dome pressure = 950 psig	Rx dome pressure = 980 psig
Core flow = 54.25 Mlb/hr	Core flow = 81 Mlb/hr
Rx power = 1660 MW	Rx power = 2490 MW
P-1 MCPR = 1.57	P-1 MCPR = 1.37

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.07 (2.00)

Water enters the regenerative heat exchanger from the reactor at 538 deg F and exits to the NRHX at 233 btu/lbm

a. If water exists the demineralizers at 120 deg F, WHAT is the temperature of the water returning to the reactor? Show all work and state all assumptions. (1.25)

b. If a (10%) leak developed downstream of the demineralizer, WHAT would be the temperature of the water returning to the reactor? (0.75)

QUESTION 5.08 (2.00)

While Nine Mile Pt-2 is operating at 90%, extraction steam to the highest pressure feedwater heater is removed. An engineer observed that the turbine load increased by 20 MW electric and concluded that this action has improved (increased) the plant's thermodynamic efficiency (not heat rate).

IS this conclusion correct? EXPLAIN your answer fully (INCLUDE WHAT caused electrical output to increase.) (2.0)

QUESTION 5.09 (3.00)

You are currently operating at 100% power BOL when you lose partial feedwater heating.

a. If the STA tells you that feedwater temperature decreased by 10 deg F, voids decreased by 2%, WHAT would be the corresponding temperature change to the fuel temperature. (Assume no rod movement, recirculation flow changes and the reactor reactivity returns to zero.) (1.5)

b. If the same situation were to occur at EOL WHAT would be the corresponding reactivity changes (MORE NEGATIVE, LESS NEGATIVE, NO CHANGE) to each of the above coefficients (i.e. delta T mod, delta (%voids), delta (T) fuel)? (1.5)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.10 (3.00)

As the reactor is taken from COLD SHUTDOWN to RATED OPERATING CONDITIONS, HOW are the following affected and WHY?

a. The MAGNITUDE of the MODERATOR TEMPERATURE COEFFICIENT (1.0)

b. DIFFERENTIAL CONTROL ROD WORTH (1.0)

c. The MAGNITUDE of the FUEL TEMPERATURE COEFFICIENT (Doppler) (1.0)

QUESTION 5.11 (2.00)

Three (3) minutes following a scram from 100% power, reactor power is 75 on IRM Range 4 and decreasing. WHAT will the indicated power be one (1) minute later? SHOW calculation and EXPLAIN any assumptions made (2.0)

(***** END OF CATEGORY 05 *****)

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

QUESTION 6.01 (3.00)

EXPLAIN WHAT affect the following failures would have on reactor level. WHY? (Assume 3-element control and Channel A controlling.) ~~(3.0)~~

a. 'C' steam line flow signal fails low (+0.75)

b. Channel 'A' reactor level detector signal fails low. (+0.75)

c. Loss of RFP lube oil to the 'A' pump servo motor TO FCV. (0.75)

d. inadvertent activation of the setpoint setdown circuitry (+0.75)

QUESTION 6.02 (2.50)

Concerning the Safety Parameter Display System

a. WHAT are the available level one (1) display(s)? (0.5)

b. WHAT are the available safety function blocks and WHAT parameters are used to determine the safety function status? (2.0)

QUESTION 6.03 (2.00)

The reactor is at 100% power with the generator synced to the grid. Electrohydraulic Control (EHC) load set is 105%. By using the attached EHC diagram, EXPLAIN WHAT would happen (control valve, bypass valve) in the following circumstances.

a. load limit potentiometer reduced to 95% (0.5)

b. maximum combined flow limit potentiometer reduced to 95% (0.5)

c. "A" pressure regulatory ~~(setpoint)~~ detector fails low (0.5)

d. failure of two (2) bypass valves full open (0.5)

ESTION 6.04 (1.00)

WHAT two (2) ~~[illegible]~~ conditions will cause the RSCS to apply a rod block to a contr[illegible] rod? (1.0)

ESTION 6.05 (2.00)

ANSWER the following questions concerning the Standby Liquid Control (SLS) System.

a. The minimum concentration needed to shutdown the reactor from rated conditions is ______________ ppm in a minimum of ______________ minutes. (0.5)

b. WHAT is the purpose(s) of the interface between the Instrument Air Systems with the SLS system. (1.0)

c. The auto start feature is interrupted by either a loss of offsite power or LOCA. (TRUE or FALSE.) (0.5)

QUESTION 6.06 (2.[illegible])

Concerning the CRD [illegible]raulic System:

a. The reactor operator is going to increase drive pressure to the HCU. WOULD you as the acting SRO direct him to OPEN or CLOSE the drive water pressure control valve? (0.5)

b. EXPLAIN HOW your action in part has changed the following flow rates (INCREASE, DECREASE, NO CHANGE) (1.5)

1. scram valve charging flow
2. CRD total system flow
3. cooling flow

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.07 (2.00)

ANSWER the following questions based upon the situation described below.

The RRCS is fully operational. The RRCS receives a reactor water low level (105 inches) signal in both complementary logics of a RRCS channel and remains in for 200 seconds. It takes 100 seconds from the initial reactor water low level signal before the APRM level is downscale.

a. Which of the four logics integrated into RRCS are actuated at T = 0 seconds? (0.5)

b. Which logics are actuated at T = 25 seconds? (0.5)

c. Which logics are actuated at T = 98 seconds? (0.5)

d. How long from T = 0 seconds is it before the RRCS can be reset? (0.5)

QUESTION 6.08 (1.50)

The Generator Gas Control System provides the main generator with hydrogen to cool the rotor windings and internal components. For this system, three (3) parameters of information (purity, pressure, and temperature) are available in the Control Room concerning generator hydrogen.

a. HOW DOES each effect generator cooling capability if deviated from normal 100% power operations? (Assume purity and pressure to decrease and temperature to increase.) (0.75)

b. You are in the process of purging the main generator with carbon dioxide. STATE HOW the following failure would effect this operation (ABORT, DELAY or NO EFFECT). (0.75)

1. pipe failure at the exit of the electric vaporizer heater

2. low level in Storage Tank (TK1)

3. low generator gas pressure (2 psig)

QUESTION 6.09 (2.50)

Both the SRM and IRM compensate their detector signals with a unique type of discrimination process.

a. Briefly DESCRIBE HOW each system (SRM/IRM) accomplish this task. (1.50)

b. WHY is there a difference between the two (2) discrimination processes? (1.00)

QUESTION 6.10 (2.00)

An automatic HPCS initiation has occurred. Subsequently HPCS injection was automatically terminated due to high reactor water level.

a. WHAT component in the HPCS system functioned to terminate the injection? (0.50)

b. Assuming no operator action, HOW WILL HPCS respond to a subsequent decreasing water level? (0.50)

c. WHAT would be the response to decreasing water level if HPCS injection has been terminated manually by closing the injection valve? (0.50)

d. If the HPCS system had switched sources from the CST to the supression pool due to low CST level and the CST level had subsequently recovered, WILL the system automatically switch back to the CST suction? (0.50)

QUESTION 6.11 (2.50)

a. WHAT are the differences in modes of operation (System Cooling Line-ups) for the RHS Loops A and B?

b. WHAT is the reason for the interlock between the

1. shutdown cooling suction valve and the test return valve?

2. pressure control valve bypass valve (MOV-23A) and pressure?

c. If a LPCI auto initiation function (high drywell) were overridden to realign the RHS system to the shutdown cooling mode and another LPCI signal (low level) were to come in, would the RHS loop realign from the shutdown cooling mode to the LPCI mode? EXPLAIN.

QUESTION 6.12 (1.00)

The plant is operating at 100% power. APRM channels A and ... have failed high. You call the I&C Technician to investigate. A Plant Auxiliary Operator wants to shift RPS B power supply to it's alternate power source for training. Would you let him? EXPLAIN WHY or WHY not. Direct your answer toward system responses instead of administrative requirements.

QUESTION 6.13 (1.00)

WHAT condition(s) do the vacuum relief lines between the drywell and suppression chamber limit/protect against?

(***** END OF CATEGORY 06 *****)

QUESTION 01 (2.00)

WHY are each of the following [illegible] [illegible]ulation System precautions necessary (i.e., what do [illegible] [illegible]revent or ensure when observed)?

a. An idle recirculation loop [illegible] [illegible] started unless the temperature differential [illegible] [illegible] pressure vessel steam space coolant [illegible] [illegible] head drain line coolant is less than or equal to [illegible] F, and [illegible] (0.5)

b. When both loops have been idle [illegible] the temperature differential between the reactor [illegible] within the idle loop to be started up and the coo[illegible] in the reactor pressure vessel is less than or [illegible] 50 deg F, or (0.5)

c. When only one loop has been idle [illegible] the temperature differential between the [illegible] within the idle and operating recirculation loops [illegible] than or equal to 50 deg F and the operating loop [illegible] less than or equal to 50% of rated loop flow [illegible] (0.5)

d. TRUE or FALSE: The operat[illegible] recircu[illegible]lation motor starts from ambient [illegible] with a required 5-minute delay [illegible] (0.5)

QUESTION 02 (2.00)

Assume a loss of Station Air [illegible]

a. WHAT three (3) automatic actions [illegible] be verified as having occurred if STATION AIR HEADER [illegible] is observed to be at [illegible] psig? (Setpoints required) (1.5)

b. Under WHAT circumstances [illegible] the Operator Actions for a Loss of Station and [illegible] require the reactor to be manually scrammed? (0.5)

OCEDURES - NORMAL, ABNORMAL, EMERGENCY AND
DIOLOGICAL CONTROL

TION 7 03 (2 00)

Concerning Radiation Work Permits (RWP)

a WHEN would an extended RWP be issued verses a standard RWP? (.5)

b WHAT is the maximum length of issuance of an extended RWP? (.5)

c You are to assign a work force to recharge the CRD accumulators Radiation protection has said that a 25 mrem/hr dose exists in the area Maintenance states that the job should take

1 4 hours with one employee, or
2 3 hours with two employees

WHICH work force would you choose for the work and WHY? (1 0)

QUESTION 7 04 (1 50)

Answer the following question concerning radiation and radiological control for a 20-year-old employee with an accumulated occupational dose of 8 rem

a WHA would be the employees maximum federal limit for th uarter? (0 5)

b COULD this employee be eligible for a life saving action and not violate any federal limits? EXPLAIN (1 0)

QUESTION 7 05 (2 50)

According to the Site Emergency Plan, the Emergency Director has certain responsibilities and/or authorities that may NOT be delegated to a subordinate during emergency conditions List these five (5) responsibilities/authorities (2.5)

QUESTION 7.09 (3.00)

USE the attached figures from N2-EOP-SPL to ANSWER the following questions.

a. DETERMINE the minimum suppression pool level given a RPV pressure of 700 psig and suppression pool temperature of 160 deg F? (1.0)

b. WHAT is the basis for the Heat Capacity Level Limit curve and WHICH area is the safe area of operation? (Above or below the line.) (1.0)

c. EXPLAIN WHAT would happe if drywell spray were initiated above the Drywell Spray Initiation Pressure Limit? (1.0)

QUESTION 7.10 (2.00)

Procedure N2-EOP-SPT (Suppression Pool Temperature Control) directs the operator to "runback recirc. and manually scram" if an SRV has beenstuck open and cannot be closed.

a. WHY is recirc. runback prior to reactor scram? (1.0)

b. Following the reactor scram you are required to depressurize the reactor, if the suppression pool temperature cannot be maintained within the Heat Capacity Temperature limit. You are also cautioned not to "depressurize the RPV below 60 psig unless motor driven pumps sufficient to maintain RPV water level are running and available." WHAT is the basis for the caution and WHAT system/components does it specifically address? (1.0)

QUESTION 7.11 (.90)

ANSWER the following question concerning the main generator and load changes. USE the attached Power Factor Chart.

You are operating at a 0.85 lagging power factor with [illegible] psig H2 and the load dispatcher orders you to drop your power factor to a 0.9 lagging power factor but maintain maximum MWe output. In general, HOW would you change your operating conditions? (INCLUDE in your answer the initial conditions (MWe, KVA), a brief discussion of the power change, and the final conditions (MWe, KVA).) (.90)

QUESTION 7.12 (.80)

According to NI [illegible] Main Turbine, there are several precautions and time limitations associated with turbine startup to assure proper [illegible] warmup, and to preclude [illegible] excessive vibration.

a. WHY should shell warming begin as soon as possible after steam seals [illegible] and WHAT might [illegible] warming is [illegible] delayed?

b. WHAT might [illegible] if [illegible] stage pressure exceed [illegible] psig during shell warming? [illegible]

QUESTION [illegible]01 (2.00)

[illegible] the following situations:

a. [illegible]ording to Technical Specifi[illegible] PERMISSIBLE to [illegible] from startup to run if IRM[illegible] are inoperable? [illegible]AIN [illegible]

b. [illegible] the same IRMs were found inoperable [illegible] in run, [illegible] violate any Technical Spe[illegible] by:

Staying in Run? EXPLAIN [illegible]

Placing the mode switch in Startup? EXPLAIN [illegible]

QUESTION [illegible] 02 (3.00)

The [illegible]vision I Diesel is operating and is [illegible] minutes into a surveillance test when the air st[illegible] system fails. The ma[illegible]ance repair team estimates [illegible] minimum repair time. (USE the attached Tech. Spec. to explain your answers.)

a. [illegible] the Diesel Generator inope[illegible] Tech. Spec.? EXPLAIN (1.0)

b. [illegible] all the Division I ECCS s[illegible] inoperable because of [illegible] Diesel Generator problem? [illegible] (1.0)

c. [illegible] at the same time the Divisi[illegible] core spray pump [illegible] service. WHAT added implica[illegible] this have on [illegible] Tech. Spec. position? (1.0)

QUESTION [illegible] 03 (2.00)

The [illegible]actor operator is performing a surveillance of the Standby Liq[illegible] Control System and due to system modifications a procedure st[illegible] becomes impossible to perform.

a. [illegible]der this condition CAN a temporary change be issued? (0.5)

b. [illegible]AT three (3) "key points" must be adhered to when issuing [illegible] temporary change? (1.5)

QUESTION 8.04 (3.00)

LIST the Nine Mile Pt. 2 Tech. Spec. Safety Limits required in Operational Condition 1. (Setpoints required.) (3.0)

QUESTION 8.05 (2.50)

A weekly surveillance, normally performed on Friday, was performed on the following days due to manpower limitations over the Thanksgiving Holiday.

Friday - November 22
Wednesday - November 27 (5 days from last surv.)
Thursday - December 5 (8 days from last surv.)
Friday - December 13 (8 days from last surv.)

a. HAVE the surveillance requirements been exceeded for this set of dates (YES/NO)? EXPLAIN your answer. (1.5)

b. WHEN is the maximum allowable date that the next surveillance can legally be performed? (INCLUDE HOW you determined this date.) (1.0)

QUESTION 8.06 (2.50)

With the reactor plant in mode 1, it is determined that four (4) gallons per minute are being collected by the Drywell floor drain system. Also, the Drywell equipment drain system indicates 22 gallons per minute (steady) collection rate.

a. WHAT are the maximum allowable plant leakage limits? (1.5)

b. STATE the actions required by Tech. Spec. for the above condition (if any). (1.0)

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8 07 (2 50)

Concerning shift complement and shift turnovers

a HOW MANY SRO, RO, STA are required in operation condition 1? (1 5)

b You come on shift and find that one (1) RO has not reported for duty CAN your crew accept shift responsibilities in this condition? WHY or WHY NOT? (0 75)

QUESTION 8 08 (1 50)

Concerning the APRM setpoints for power distribution limits

a CALCULATE the scram trip setpoint(s) if the reactor is operating at 3000 MW TH with most limiting LHGR mode operating at 10 KW/ft ASSUME an LHGR limit of 13 4 KW/ft (0 75)

b DOES this result require any APRM adjustment? WHY or WHY NOT? (0 75)

QUESTION 8 09 (1 50)

Technical Specification 3 7 1 1 requires two plant service water pumps per loop to be operable and provides explicit action requirements if one service water pump per loop is inoperable If both of the service water pumps per loop were to become inoperable, no specific action statement would apply

a WHAT would be your required action? (1 0)

b HOW SHOULD an operator interpret tech specs in this instance and in other similar instances not directly provided for in the action statements to insure the intent of the specifications are met? (1 5)

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.10 (2.00)

Using the attached Technical Specifications, DETERMINE the maximum time that the reactor may continue operation given the following malfunctions. Reference the sections of tech specs used in determining your answer.

MOVE 1.

a. It is discovered that valve F048A (RHR "A" Heat Exchanger Bypass) is failed open and cannot be closed. (1.0)

b. Susequent to the malfunction in (a) above, it is found that RHR pump B is inoperable. (1.0)

QUESTION 8.11 (2.50)

The RCIC outboard isolation valve (2ICS-MOV121) motor controller has failed in the deenergied position and the valve won't shut. Maintenance is currently attending to the problem. By using the attached Technical Specifications:

a. STATE which Tech. Specs. apply to this problem. (0.5)

b. STATE whether RCIC is OPERABLE or INOPERABLE and GIVE ANY necessary action statement(s) required. (2.0)

(***** END OF CATEGORY 08 *****)
(************* END OF EXAMINATION ***************)

ANSWERS -- NINE MILE POINT . -85/12/10-C A SLY

ANSWER 5 01 (2 00)

a FALSE (+0 5)
b FALSE (+0 5)
c TRUE (+0 5)
d FALSE (+0 5)

REFERENCE
1 NMP-2 Operations Technology, Module 1, Part 15 pp 1-15-1, I-15-2, Student Learning Objectives No 2, 3

ANSWER 5 02 (2 00)

a increase (+0 5)

b decrease (+0 5)

c decrease (+0 5)

d [illegible] ~~increase~~ (+0 5)

REFERENCE
1 NMP-2 Operations Technology, Module 1, Part 14 pp 1-14-9, I-14-10 Student Learning Objective No 4
[illegible]

ANSWER 5 03 (2 00)

Supercritical (+0 5) When the period reaches infinity, the reactor is exactly critical on prompt neutrons (+0 5) After the rod insertion stops the delayed neutron precursors which were formed in previous generations and at a higher power level tend to pull power back up (+0 5) Therefore, the reactor is still supercritical due to the latent effect of delayed neutrons (+0 5) (+2 0 Total)

REFERENCE
1 NMP-2 Operations Technology, Module 1, Part 11 pp I-11 x

ANSWER 5 04 (2 50)

a See Figure 8-11 in Reference material
criticality predicted at pin 16
(+0 5 for plot and +0 5 for usage)

b next reading = 1/2(16 - 6) + 6 = after the 11th fuel bundle
therefore 5 more fuel bundles may be loaded (1.0)
<no double jeopardy>

c away from the detector (0.5)

REFERENCE

1 NMP-2 Operations Technology, Module 1, Part 8, pp 1-8-10 and 1-8-13, Figure 8-11, Student Learning Objective No 4

ANSWER 5 05 (2 50)

a The lowest pressure that the tank could drop to would be the saturation pressure for 60 deg F which is 0 256 psia (+0 5)
<5 ft head or 2.17 psia OR 4.41 ft H_2O + vapor press = 5 ft H_2O ...

b Assuming head loss due to flow negligible, the answer is no Cavitation would not begin until the level drops below 5 ft in the tank (+1 0)

c Yes (0.5) The added pressure of 14.7 psia at the pump suction would allow all of the water to be removed (+0 5)

REFERENCE

1 NMP-2, SLO for Fluid Statics, Dynamics and Delivery, No 10, pp 15, 16

ANSWER 5.06 (2.00)

Assuming 100% core flow is 108.5 Mlb/hr (+0.25), min MCPR (limit) = 1.24 (+0.5)

For Condition 1
% core flow = 54.25/108.5 = 50%
from Figure 3.2.3-1 Kf = 1.175 (+0.25)
therefore the MPCR(limit) = 1.24(1.175) = 1.457
delta(MCPR) = 1.57 - 1.457 = ~~1.13~~ (.113)

Check calc

For Condition 2
% core flow = 80/108.5 = 74.6%

Figure 3.2.3-1 Kf = ~~1.005~~ 1.05 (+0.25)
therefore the MCPR(limit) = 1.24(1.005) = 1.25
delta (MCPR) = 1.37 - 1.25 = ~~1.20~~ (.068)
Condition 2 is closer to limits (+0.5)
(0.25 for math)

REFERENCE

1. NMP-2 Student Learning Objective for BWR Thermodynamics and Thermal Hydraulic Limits, No. 7, p. 4

2. General Electric Thermodynamics, Heat Transfer, and Fluid Flow, MTC, March 1983, pp. 9-96 to 9-99

3. NMP-2 Tech. Specifications 3/4.2.3, Minimum Critical Power Ratio, Figure 3.2.3-1 and Table B2.1.2-2

ANSWER 5.07 (2.00)

a. 1. Set up equation
   A. MCp delta(T) = M delta(h)
   B. Cancel M because both are equal. Cp delta(T) = delta (h)
   C. Lookup h(f) for 538 deg F = ~~543~~ Btu/lbm (+0.25) 534.2
      (+0.5 pts for equation and logic)

   2. Solve for delta (h) 233 Btu/lbm = 264 °F
      A. ~~(543~~-233) Btu/lbm
      B. delta (h) = ~~310~~ 301 Btu/lbm

   3. Solve for T(hot) to reactor h_f = 4297
      Cp (T(hot) - T(cold)) = delta (h)
      T(hot) = T(cold) + [delta (h) / Cp] 388 Btu/lbm ⇒ 413°F
      T(hot) = ~~420~~ deg F (0.5) ⇒ T_{hot} = 394

b. Solve for T(hot) to reactor
   1. M2 * Cp * (T(hot)-T(cold)) = M1 * delta (h)
   2. M2 = 90% M1
   3. T(hot) = T(cold) + [delta (h) / (0.9 * Cp)]
   4. T(hot) = 464 deg F (+0.5)

REFERENCE
1. Thermodynamics Lesson Plan, Heat Transfer and Heat Transfer Equipment, p. 13 of 13

2. NMP-2 Examination Bank Category 1.5, pp. 67, 68, Student Learning Objective No. 3

ANSWER 5.08 (2.00)

No (+0.25) thermo efficiency is a comparison of Energy In to Energy Out (+0.5). The increase in output results from no steam being diverted to the high pressure feedwater heater (+0.5). Because the feedwater is now cooler, more energy from the reactor is required to bring the water up to saturation temperature (+0.5) thus thermo efficiency is down (+0.25). (+2.0 Total)

REFERENCE
1. NMP-2 Power Plant Cycles, pp. 5-7, Student Learning Objective No. 1

2. General Electric Thermodynamics, Heat Transfer, and Fluid Flow, MTC, March 1983, pp. 6-38, 6-66

ANSWERS -- NINE MILE POINT 2 -85/12/10-G A SLY

ANSWER 5.09 (3.00)

alpha(d)(delta(T))fuel = alpha(m)(delta(T))mod + alpha(v)delta(%V)
(0.25pts for equation)

alpha(m) = -1 x 10E-4 delta K/K/deg-F (0.25)

alpha(d) = -1.2 x 10E-5 delta K/K/deg-F (0.25)

alpha(v) = -1.0 x 10E-3 delta K/K/%V (0.25)

delta(T) =[(-1 x 10E-4)(-10) + (-1 x 10E-3)(-2)] / (-1.2 x 10E-5)
delta(T)fuel = -250 deg F or 250 deg F increase in fuel temperature (+0.5 pts) 300°F if 1.0x10^-3

b. delta(ro) = alpha(x) * delta(X)
delta(ro)dop = more negative (+0.5)
delta(ro)void = less negative (+0.5)
delta(ro)mod = less negative (+0.5)

REFERENCE
1. NMP-2, Operations Technology, Module I, Part 12, pp. 12.5, 12.7, Fig. 12-6, 12-7. Student Learning Objectives No. 2 & 3
2. NMP-2, Operations Technology, Module I, Part 13, pp. 13.5, 13.6. Student Learning Objectives No. 2 & 3

ANSWER 5.10 (3.00)

a. INCREASES [+0.25] Because the change in density of water per degree F change in temperature increases with increasing temperature [+0.75] (+1.0 TOTAL)

b. INCREASES [+0.25] Because neutron leakage from the fuel cell to the volume around the control rod increases exposing the rod to a higher thermal neutron flux [+0.75] (+1.0 TOTAL)

c. DECREASES [+0.25] Because the amount of resonance broadening per degree F change fuel temperature decreases OR at higher fuel temperatures most of the broadening takes place at the higher energies where fewer and fewer neutrons exist [+0.75] (Either reason correct for full credit.) (1.0)

REFERENCE

1. NMP2, Operations Technology, Module I, Part 12,13,14 pp. I-12-3, I-12-4, I-12,5, I-13-2, I-13-3, I-14-6, I-14-7, Student Learning Objective No. 12-2, 13-2, 14-4a.

ANSWER 5.11 (2.00)

Using P = Po e ** (t/T) (+0.5)
= 75 e ** (-60/80) (+0.25)
= 35 on Range 4 (+0.25)

On a down power transient, with large negative reactivity insertions, ~~[illegible]~~ the stable dacay period is determined by the longest lived half-life (+0.5) For this example it is assumed to be -80 sec. (+0.5)

REFERENCE

1. NMP2, Operations Technology, Module I, Part 10, p. I-10-2 Student Learning Objective No. 3

ANSWER 6.01 (3.00)

a. decrease (+0.25) due to steam/feed mismatch requiring less water (+0.5) (+0.75 Total)

b. increase (+0.25) due to level mismatch requesting more water (+0.5) (+0.75 Total)

c. no change (+0.25), servo would lock up valve as is (+0.5) (+0.75 Total)

d. decrease (+0.25), due to reduction in operator setpoint of one-half input value (+0.5) (+0.75 Total)
185" is input setting level setpoint

REFERENCE

1. NMP-2 Operations Technology, Module IX, Part 6, pp. 4, 5 Student Learning Ojbective Nos. 4, 7
2. NMP-2, IOP-7, pp. 3

ANSWER 6.02 (2.50)

a. Safety function status display (+0.5) OR

APRM flux
Core flow
Rx pressure
RPV level
Drywell ...
(0.1 each)

b. 1. reactivity control - APRM status

2. core cooling - RPV level

3. coolant system integrity - RPV pressure or drywell pressure

4. containment integrity - drywell pressure, drywell oxygen concentration, or suppression pool temperature (0.25 each display, +0.25 for each status parameter)

REFERENCE

1. NMP-2 Operations Technology, Module VI, Part 12, SPDS, pp 2, 3, 4 of 8, Student Learning Objective Nos. 2, 1

ANSWER 6 03 (2.00)

a. control valves close 5% (+0 25), open one bypass valve (+0 25) (or similar answer on diagram)

b. Control valves close 5%, (0 25) (reactor scram probable due to increasing pressure since) bypass valves will not be open (+0 25)

c. (will develop a pressure error of 800 psid) This will be a demand for maximum opening of all valves However, due to the action of the maximum combined flow limiter, control valves will go to ~~105%~~ 100 demand and bypass valves to ~~10%~~ Throttle

This will depend on max combined flow set

d. control valves close to 90% (+0 25) to maintain ~~Rx~~ pressure at 920 psig (+0 25)

REFERENCE

1. NMP-2 Operations Technology, EHC, Rev 1, pp 2, 5 to 9 of 14, Student Learning Objective Nos 5, 6, 8, including EHC Figure 3

ANSWER 6 04 (1 00)

a. If substitute position data has already been entered from the RSCS operators panel, that rod has been moved one notch, and good position data is still missing, then a rod motion insert and withdraw block will occur (+0 5)

b. From 75% rod density to the LPSP, only notch rod movement is allowed between 00 and 12 (+0 5)

REFERENCE

1. NMP2 Operation Technology, Module VI, Part 6, p 11 SLO 5 a

add com... rod not in sequence selected
... Rod at insert bank limit
... Rod at withdraw bank limit
... Rod data fault
RSCS inop

any ... for full credit

ANSWERS -- NINE MILE POINT 2 -85/12/10-G A SLY

ANSWER 6.05 (2.00)

a 660 ppm (+0.25), 50 (+0.25)

b Instrument Air - air to bubbler level indicator (+0.5)
sparging air for preparation of poison solution (+0.5)

c FALSE (+0.5)

REFERENCE
1 NMP-2 Operations Technology, Module VI, Part 9, pp 2, 5, 8, 9, Student Learning Objective Nos. 2, 3, 4, 5

ANSWER 6.06 (2.00)

a close (+0.5)

b 1 no change (+0.5)
2 no change (+0.5)
3 ~~decrease~~ (+0.5)
no change

REFERENCE
1 NMP-2 Operations Technology, Module III, Part 5, pp 4, 5, 6, Student Learning Objective No 4

ANSWER 6.07 (2.00)

Alternate Rod Insertion, Recirculation Pump Trip
b None
c Standby Liquid Control [illegible] ISOLATION 20 sec. ARI
d ~~30 seconds~~ 11 min 48 sec. 10 min + 48 sec for
(+0.5 for each, +2.0 TOTAL)

REFERENCE
1 NMP-2, Operations Technology, Module VI, Part 8, pp 2, 4, 7
Student Learning Objectives No 4

ANSWER 6.08 (1.50)

a. 1. Purity - as purity decreases, cooling capability decreases (+0.25)

2. Pressure - as pressure decreases, cooling capability decreases (+0.25)

3. Temperature - as temperature increases, cooling capability decreases (+0.25)

auto isolate on low level on tank

b. 1. Terminate purge due to single purge line (+0.25), (generator isolates)

auto isolate

2. ~~no effect~~ (Tank 1 isolates) (+0.25),(TK2 ~~starts~~ not 8 supplies)

3. no effect (+0.25), (dump valve not opened or used during purge)

REFERENCE

1. NMP-2 Operations Technology, Module ~~VIII~~, Part 6, pp. 4, 6, 8, 10, Student Learning Objective Nos. 2, 4, 5

2. Step 6.1.2

ANSWER 6.09 (2.50)

a. Pulse height - neutron pulse larger than gamma pulse (+0.25), pulse height discriminator (+0.25) chops gamma and only passes neutron pulses (+0.25)

Cambelling - neutron pulse is larger than gamma pulses (+0.25), Cambelling (+0.25) squares the two signals then chops gamma and passes only neutrons (+0.25) γ becomes insignificant

(+1.5 Total)

b. Due to the low number of events and greater sensitivity (+0.25), the SRM deals with individual counts (pulses) (+0.25) where the IRM deals with time averaged signals (+0.5) (+1.0 Total)

REFERENCE

1. NMP-2 Operations Technology, Module VI, Part 1, SRM, pp. 2, 8, Student Learning Objective No. 3

2. NMP-2 Operations Technology, Module VI, Part 2, IRM, pp. 3, 4, Student Learning Objective No. 3

ANSWERS -- NINE MILE POINT 2 -86/12/10-G.A. SLY

ANSWER 6.10 (2.00)

a. closure of the HPCS injection valve (MOV-107) (+0.5)

b. ~~auto restart~~ on the low-low setpoint (+0.5)

c. stay in manual bypass and not reinitiate (+0.5)

d. no (+0.5)

REFERENCE
1 NMP-2 Operations Technology, Module IV, Part 2, HPCS, pp. 1, 2, 5, 6, 7, Student Learning Objective Nos. 2, 4, 7

ANSWER 6.11 (2.50)

a. RHS 'B' - Head Spray Mode (+0.25)
- Containment Flooding Mode (+0.25) or - (S.C.

b. 1. Prevent inadvertent draining of the vessel (+0.5)
2. Prevent exceeding RHS design pressure (+0.5)

c. No The ... valve MOV 1 will be shut in SD cooling
~~Yes~~ (+0.25), ~~because the second LPCI initiation signal will realign the system by reopening the LPCI injection valve~~ (+0.75)
No auto action with MOV-1 (INJECTION WON'T Reopen)

REFERENCE
1. NMP-2, Operations Technology, Module IV, Part 5, RHS, pp. 5, 9, 10. Student Learning Objectives No. 1, 5, 6
N2-IOP ...

ANSWER 6.12 (1.00)

No (+0.25) When transferring RPS power supplies, the RPS is momentarily deenergized because the transfer is break before make. This would result in a scram due to the 1/2 scram already present (+0.75)

REFERENCE
NMP2, N2-IOP-97, RPS, p. 6

ANSWER 6.13 (1.00)

-or- Limit floor ΔP to < 3 psid - (+0.5)

Limits negative pressure differential (+0.5) to prevent drawing water up the downcomer from the suppression pool to the drywell. (+0.5) add floor →

REFERENCE

1. NMP-2 Operations Technology, Primary Containment, Rev. 1, p. 6

T.S. Bases B-3/4-6.4

ANSWERS -- NINE MILE POINT 2 -85/12/10-G A SLY

ANSWER 7.0 (2.00)

a. Prevent undue stress on vessel

b. Prevent undue thermal shock in recirculation pump and nozzles

c. Prevent undue thermal stress on vessel nozzles and bottom head.

d. False.

REFERENCE
1. Technical Specifications 3/4.4.1.4 and B 3/4.4-1
2. NMP-2, N2-IOP-29, Recirculation, p. 3

ANSWER 7.02 (2.00)

alternate answers due to conflicting setpoints in procedure

a. 1. Lag
~~Second~~ station air compressor on standby has started at 100 psig (+0.5) (or 90 psig)

2. Standby or backup
Third station air compressor on standby has started at 90 psig (+0.5) (or 85 psig)

3. Unit Service air
Station air isolation valve (AIAS-AOV171 ~~PSO-F401~~) closed at 85 psig (+0.5) ~~(or 85 psig)~~

b. If air header ~~[illegible]~~ pressure reaches 60 psig / or rapid air loss. (+0.5)

REFERENCE
NMP-2, N2-IOP-19, Instrument Air, pp. 6, 8, 9, 13

ANSWER 7.03 (2.00)

a. routine or repetitive work (+0.5)
b. 1 year (+0.5)
c. group ~~2~~ 1 (+0.5) due to ALARA program (+0.5)

REFERENCE
1. NMP-2, S-RP-2, RWP Procedure, p. 14
2. NMP-2, S-RP-7, ALARA, pp. 2,3 Reg Guide, 8.29-9, Item 6.

ANSWERS -- NINE MILE POINT 2 -85/12/10-C A SLY

ANSWER 7.04 (1.50)

a. 2 re (+0.5) due to 5(N-18)
b. no (+0.5), exceeds 5(N-18) limit (+0.5) - or yes, once in a lifetime

REFERENCE
1. NMP-2, S-RP-1, Access and Radiological Control, pp. 1, 12, 17
2. NMP-2, EPP-15, Health Physics Procedure, p. 3

ANSWER 7.05 (2.50)

1. Making decision to notify offsite emergency management agencies

2. Making protective action recommendations as necessary to offsite emergency management agencies

3. Classification of the emergency event

4. Determining the necessity for a site evacuation

5. Authorizing emergency workers to exceed normal radiation exposure limits

(+0.5 each)

REFERENCE
NMP-2, SEP Sec. 5, Organizational Control of Emergencies, pp. 4, 5

ANSWER 7.06 (2.00)
or natural circulation (+0.5)

a. Concentrate boron (+0.5) enhance void generation (+0.5)

b. Max. temp. at which SLC initiation will result in injection of hot shutdown boron weight before the supp. pool reaches the HCTL in an ATWS. (i.e. assures shutdown prior to emergency depressurization) (+1.0)

REFERENCE
1. NMP-2, N2-EOP-C7, Level/Power Control, p. 9, Student Learning Objective Nos. 1, 3

2. NMP-2, N2-EOP-RO, RPV Reactivity Control, p. 2, 12 of 21

ANSWERS -- NINE MILE POINT 2 -85/12/10-C A BLY

Student Learning Objective No 3

ANSWER 7 07 (1 50)

Following flooding of vessel (+0 25) the flow path would be

(Main steam lines) to suppression pool via SRVs (+0 25)

Suppression pool to vessel via core spray (+0 25) or LPCI (+0 25) <or ECCS pumps>

Heat is removed from suppression pool by suppression pool cooling mode of RHR (+0 5)

REFERENCE
NMP-2, N2-EOP-C5, Alternate Shutdown Cooling, p 1

ANSWER 7 08 (2 50)

alt answer
For reactor level control so as not to flood Rx vessel or use Rx drain to maintain level

a. The CRD pump will increase water level (+0 25) and there is no outlet flow path established (+0 25)

b. Because cooling flow is lost to the regeneratives heat exchanger (+0 25) increasing the outlet temperature to the NRHX (+0 25), possibly causing isolation of system (+0 5)

Hot shutdown? Hot shutdown mode?

c. Hot shutdown with no recirculation pumps operating (+0 5) minimizes thermal stratification of vessel water (+0 5)
or HoT Standby, for water quality, or Rx water level control

REFERENCE
NMP2, Operations Technology, [illegible] , pg 8 of 12
N2-IOP-37 pg 4 Caution #1

ANSWER 7 09 (3 00)

a CAF (+1 0 TOTAL) 191°

b Above (+0 [illegible]) assures [illegible]icient heat capacity available to absorb the energy from [illegible] blowdown (+0 5)

c Spray initiation above this lim t may result in a containment depressurization rate that exceeds the relief capacity of the drywell ~~and [illegible]~~ or vacuum breaker (+1 0 TOTAL)

ANSWERS -- NINE MILE POINT 2 -85/12/10-C A SLY

REFERENCE
1 NMP-2 N2-EOP-~~PCP~~ RC, ~~p. 2, 8~~ and p. 8, ~~10, 16~~ of 18
Student Learning Objective No. 3

ANSWER 7.10 (2.00)

a. To minimize the transient (+0.5)

b. RCIC (+0.5) will isolate at 50 psig (+0.5) and you want assurances that you have an injection mode available prior to depressurization (+0.5)

REFERENCE
~~1 NMP-2 N2-IOP-04, SRV/ADS, p. 7~~ NMP2, N2-EOP-RC, pg. 10 of 21.

2 NMP-2 N2-EOP-SPT, p. 2, 3 and p. 8, 10 of 13
Student Learning Objective No. 3

ANSWER 7.11 (2.00)

Initial State
~~0.875~~ 1,275 MWe and ~~400~~ 425 KVAR (+0.25)

You should reduce generator load by recirc or rods to 1.210 MWe (+0.25), then raise reactive load (VAR) by adjusting the AC voltage regulator (+0.25) (+0.5 for order of steps)

Final State
1.210 MWe and 600 KVAR (+0.25)

REFERENCE
1 NMP-2, N2-IOP-68, Main Gen., p. 5 and figure 3
Power Factor Chart Provided

ANSWER 7.12 (2.00)

a. This is necessary to prevent uneven heating of the rotor (+0.5). If it is not started, a rotor bow condition could result (+0.5)

b. The setpoint of the "Turbine Stop and Control Valve Closure bypassed" annunciator could be exceeded (+0.5) and a reactor scram would result (+0.5)

REFERENCE
1. NMP-2 N2-IOP-21, Precautions 2, 3

ANSWER 8.01 (2.00)

a. Yes (+0.5), following putting RPS trip System A in the tripped position (+0.5) as per 3.3.1.a (+1.0 Total)

b. 1. No (+0.25), IRMs are not required in Condition 1 and you may stay there (+0.25) (+0.5 Total)

2. Yes (+0.25), unless you had the RPS trip System A in the tripped position (+0.25). Specification 3.0.5 is not applicable (+0.5 Total)

REFERENCE

1. Tech. Specs, pp. 3/4 0-1, 3-1 to 3-4

ANSWER 8.02 (3.00)

a. Yes (+0.5), due to failure of surveillance 4.8.1.1.2.7 air pressure greater than 225 psig (+0.5) (+1.0 Total)
-or- a redundant systems are not operable

b. No (+0.5), due to Specification 3.0.3 which states you can be without emergency power source if you have everything else (+0.5) (+1.0 Total) → Also will get answer of T.S. 3.8.1.1 action e

c. You would be in violation of Specification 3.0.3 (+0.5), and must perform the action statement (+0.5) (+1.0 Total)

T.S. 3.8.1.1.b diesel action 7 days + [illegible] + [illegible]

REFERENCE

1. Tech. Spec., pp. 3/4 0-1, 8-1 to 8-8

-or- T.S. 3.5.1.a LPCI [illegible] action

T.S. 3.3.2.a [illegible] [illegible] [illegible]

-or- T.S. 3.5.1.e [illegible]

NSWER 8.03 (2.00)

a. Yes (+0.5)

b. 1. The intent of the original procedure is not altered (+0.5)

2. The change is approved by two (2) members of the plant management staff, at least one (1) of whom holds a Senior Reactor Operators License on the unit affected (+0.5)

3. The change is documented, reviewed, and approved by the General Superintendent Nuclear Generation or designee within 14 days of implementation (+0.5)

REFERENCE

1. NMP-2 Tech. Spec., Administrative Procedures 6.8.3

2. NMP-2 Exam Bank

ANSWER 8.04 (3.00)

1. THERMAL POWER, Low Pressure or Low Flow

Thermal Power shall not exceed 25% of Rated Thermal Power with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow (+1.0)

2. THERMAL POWER, High Pressure and High Flow

The Minimum Critical Power Ratio (MCPR) shall not be less than 1.06 with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow (+1.0)

3. REACTOR COOLANT SYSTEM PRESSURE

The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig (+1.0)

REFERENCE

1. NMP-2 Tech. Spec., pp. 2-1, 2-2

ANSWER 8 05 (2 50)

a No (+0 5), allowed to exceed weekly by 25% ~~or one day~~ (+0 25) no restriction doing them early (+0 25) Also did not exceed 3 25 times interval for three (3) consecutive surveillance (+0 5)

b Next surveillance would be Wednesday, December 19 (+0 5), because you are limited by the three (3) consecutive interval limit (22 days) from November 27 (+0 5)

REFERENCE
1 NMP-2 Tech Spec , pp 3/4 0-2

ANSWER 8 06 (2 50)

a 1 No known Pressure Boundary Leakage (+0 5)

2 5 gpm unidentified leakage (+0 5)

3 25 gpm total leakage averaged over a 24-hour period (+0 5)

b 4 gpm --- unidentified leakage

22 gpm --- identified leakage

26 gpm total leakage (+0 5)

Reduce the total leakage rate to less than 25 gpm within 4 hours or be in at least hot shutdown in 12 hours and cold shutdown in the following 24 hours (+0 5)

REFERENCE
1 NMP-2 Tech Spec , LCO, Reactor Coolant System, Operational Leakage 3.4.3.2

ANSWER 8 07 (2 50)

a SRO - 1 (+0 5)
RO - 2 (+0 5)
STA - 1 ~~and 2 Unlicensed operators~~ (+0 5)

b No (+0 25), the 2 hour exception does not apply during shift changes (+0 5) (+0 75 Total)

Yes if on shift operator stays over

.MP-2 Tech. Spec., 3 6, p 6-1 and Table 6 2 2-1

..SWER 8.08 (1.50)

a. T = FRTP/CMFLPD Both items defined in Tech. Spec. Definitions (+0.25)

T = [(300/3323)/10]/13 4 = 0.902/0.746 = 1.2 (+0.25)

S is less than or equal to (0.66 W + 51%) (+0.25)

b. No (+0.25) The Tech. Spec. require an APRM adjustment only if Tau is less than or equal to 1. (+0.5)

..EFERENCE

1. NMP-2 Tech. Spec., 3/4 2.2, LCO, Power Distribution Limits, APRM Setpoints, p. 2-5

2. NMP-2 Exam Bank

..SWER 8.09 (1.50)

a. Restore on pump within 72 hours or be in hot shutdown in 12 hours and cold shutdown within 24 hours. Also take ACTION required by Spec. ~~3.5.2 and 3.8.1.2~~ Spec. 3.0.4 (~~+1.0~~) 0.75 0.75

b. T.S. 3.0.4 delineates the measures to be taken for those circumstances not directly provided for in the action statements and whose occurrence would violate the intent of the specification (~~+1.50~~ TOTAL) +0.75

..EFERENCE

1. NMP2, T.S. bases 3.0.4, 3.5.2, 3.7.1.1, 3.8.1.2

..SWER 8.10 (2.00)

a. Restore within 72 hours or be in Hot Shutdown in 12 hrs

T.S. 3.6.2.3.a (1.0)

b. Be in at least HOT Shutdown in 12 hrs. T.S. 3.6.2.3.c (1.0)

REFERENCE
1 NMP2, T S 3 6 2 3

ANSWER 8 11 (2 50)

a T S 3 6 3 ~~and 3 7 4~~ (+0 5)

b ~~No~~ operable (+0 5) RCIC can provide its intended function, but you have violated Primary Containment Integrity requirements and must (+0 25)

1 demonstrate the inboard isolation valve operable and (+0 25)

2 within 4 hours (+0 25)

a restore the inop valve to operable (+0 25)
b isolate line (this makes RCIC inop) (+0 25)

3 or be in Hot S/D in 12 hrs and Cold S/D in 24 hrs (+0 25)

REFERENCE
1 NMP2 T S 3 6 3 and 3 7 4

ESTION	VALUE	REFERENCE
05.01	2.00	SLY0000017
05.02	2.00	SLY0000021
05.03	2.00	SLY0000022
05.04	2.50	SLY0000023
05.05	2.50	SLY0000024
05.06	2.00	SLY0000025
05.07	2.00	SLY0000026
05.08	2.00	SLY0000027
05.09	3.00	SLY0000078
05.10	3.00	SLY0000106
05.11	2.00	SLY0000111

	25.00	
06.01	3.00	SLY0000028
06.02	2.50	SLY0000029
06.03	2.00	SLY0000030
06.04	1.00	SLY0000031
06.05	2.00	SLY0000032
06.06	2.00	SLY0000033
06.07	2.00	SLY0000034
06.08	1.50	SLY0000035
06.09	2.50	SLY0000036
06.10	2.00	SLY0000037
06.11	2.50	SLY0000051
06.12	1.00	SLY0000090
06.13	1.00	SLY0000116

	25.00	
07.01	2.00	SLY0000091
07.02	2.00	SLY0000092
07.03	2.00	SLY0000093
07.04	1.50	SLY0000094
07.05	2.50	SLY0000095
07.06	2.00	SLY0000098
07.07	1.50	SLY0000099
07.08	2.50	SLY0000100
07.09	3.00	SLY0000109
07.10	2.00	SLY0000110
07.11	2.00	SLY0000120
07.12	2.00	SLY0000121

	25.00	
08.01	2.00	SLY0000080
08.02	3.00	SLY0000082
08.03	2.00	SLY0000083
08.04	3.00	SLY0000084
08.05	2.50	SLY0000086

STION	VALUE	REFERENCE
8 06	2 50	SLY0000087
8 07	2 50	SLY0000088
8 08	1 50	SLY0000089
8 09	1 50	SLY0000107
8 10	2 00	SLY0000108
8 11	2 50	SLY0000112
	25 00	
	100 00	

Battelle

375-3765

March 7, 1986

Ms. Lynn Kolonauski
U.S. Nuclear Regulatory Commission
Operator Licensing Section/Region I
631 Park Avenue
King of Prussia, PA 19406

Dear Ms. Kolonauski:

The following is the follow-up to the transmittal of the Nine Mile Point-2 results on January 24, 1986.

Enclosed is a summary of the resolutions to the unresolved comments on both the RO and SRO written examination given on December 17, 1986. Only those comments left unresolved at the time of leaving the site are included in the letter and all comments were incorporated in the Master Examinations keys. If you have any questions please call me on (509) 375-3765.

Sincerely,

G. A. Sly
EWR Coordinator

/c
Enc

RO COMMENTS:

All facility comments were either minor (i.e., typo) in nature or resolved during review of the SRO exam. Those comments which were unresolved in the SRO exam will be referenced there. All corrections and resolved comments have been incorporated into the Master Examination. One RO answer was modified during grading.

Question 1.04

Old Answer: a. power higher
flow higher

b. power lower
flow lower

New Answer: a. power higher
flow lower

b. power lower
flow higher

Reason: Since the orificing of the bundles is fixed, the same amount of potential flow would exist at the inlet of each bundle and only 2-phase flow would affect the bundle flow. This then leads to the new answer as being correct. The answer key reflects this change.

SRO COMMENTS:

Only those facility comments left unresolved or major in content are included in this list. All corrections and resolved comments have been incorporated in the Master Exam.

Question 6.11.c.(c.)

Comment – Some students were told "paths" meant "physical differences." Therefore, answer should also include:

1. RCS's to radwaste system
2. Service water to the containment spray (flooding)

Ref. NZ-[illegible]-21, p. 13, Shutdown cooling, p. 26 Service water injection into the RPV containment.

. Answer in answer key is incorrect.

No, RHR does not realign because the system is in the Shutdown Cooling Mode. The Suppression Pool suction (MOV-1) and SD suction valves (MOV-2) are interlocked (MOV-1 does not have shut signal on LPCI injection) and the suction will not shift.

Also, the question stated the injection valve was manually closed (overridden) therefore, it will not automatically reopen without resetting the initiation logic.

Ret.: RHS Ops Tech, page 9 of 17.

Resolution: Comment accepted. The answer key was changed to reflect the proper answer.

Question 7.02 c.

Comment: Answer could be 1.

Group 1 Ref: Reg. Guide, 8.29-9, Item 6
Group 2 Ref: RP-7, ALARA, p. 7.3.

Resolution: Following a check of the Regulatory Guide, 8.29-9, Item 6, Group (1) was deemed to be the correct answer.

Question [illegible]

Comment: Answer is incorrect. Answer should be:

a. "Reduces natural circulation driving head, thereby reducing core flow, reducing power which reduces rate at which heat is rejected to suppression pool."

b. Answer should stop at "[illegible]". Rest of answer is not in reference [illegible].

Ref: [illegible], [illegible] Lesson Plan, p. 12 of 21.

Resolution: Comment was accepted. Answer key was changed to reflect correct answer.

Question [illegible]

Comment

a. There is no precaution as stated in question in our procedure or Ops. Tech. material.

Acceptable answer should include either:

1. Lose reactor water level control due to either [illegible] injection or vessel [illegible].

or

c. For rejecting water to the condenser or Radwaste System to prevent use of Reactor Drain to Drywell Equipment Drain Tank.

Ref: NMP, Ops. Tech. Rmts. p. [illegible] of [illegible].

Resolution: Only Part [illegible] of Part a. was accepted as being correct.

Question [illegible]

Comment

1. There are 3 pumps per loop in service water system, therefore, "both" does not apply.

   If student interpreted this question to be 1 pump loop is inop, no action would be required because a loop is defined as 2 operable pumps ([illegible]).

2. The question stated that there was no specific action statement in Tech. Specs., however, the answer key referenced a Tech. Spec. action statement.

3. Actions in answer key would be stated if student determined [illegible] pumps loop were inoperable, then question statement of no specific action statement would apply is false.

Resolution: The facility comment was determined to have some merit, even though two [illegible] announcements of clarification were made during the written exam. For this reason the answer was modified to grade for concept only. This question should be reworded in the future to more specifically address the required knowledge. The answer key reflects the proper response for full credit.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OCT 2 8 1983

MEMORANDUM FOR: Darrell G. Eisenhut, Director
Division of Licensing

THRU: Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

FROM: Mary F. Haughey, Project Manager
Licensing Branch No. 2
Division of Licensing

SUBJECT: NRR INPUT TO SALP - NINE MILE POINT 2

The NRR SALP input for Nine Mile Point Nuclear Station, Unit 2 covering the period 10/01/82 - 09/30/83 is provided as the attachment to this memo. The findings in the report are based on comments from the project manager and reviewers who have had interactions with the licensee. No additional comments were received from NRR Division Directors in review of the draft input.

Mary F. Haughey
Mary F. Haughey, Project Manager
Licensing Branch No. 2
Division of Licensing

Attachment:
As stated

cc: Region I

 B-1

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CF

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K/11

Licensing

Significant licensing activities during this period include:

- Operating license application and acceptance review
- Completion of the revetment ditch review
- Caseload Forecast Panel, February 22-24, 1983
- Responses to acceptance review requests for information
- Management meeting on April 26, 1983
- Meeting on deviations from the Standard Review Plan, September 1, 1983
- A number of meetings and conference calls to discuss technical issues related to the safety and environmental reviews.
- Environmental Site Visit August 1-2, 1983

a. Management Involvement and Control in Assuring Quality

[redacted]

b. Approach to Resolution of Technical Issues

Major activity in this area has been the FSAR tendered in February 1983, and docketed in April 1983; [redacted]

c. <u>Responsiveness to NRR Initiatives</u>

The major items requiring response during this period were the requests for information included in the acceptance package. Responses were submitted for these requests in keeping with the requested schedule. [redacted]

d. <u>Staffing</u>

There has not been a significant opportunity to evaluate this area within the licensing area, but, in areas such as the environmental site visit and the geology and seismology meeting, [redacted]

e. <u>Safeguards</u>

[redacted]

<u>Conclusion</u>

[redacted]

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 1986

Docket No. 50-410

MEMORANDUM FOR: Richard W. Starostecki, Director
Division of Project and Resident Programs

THRU: Wayne Houston, Deputy Director
Division of BWR Licensing

FROM: Mary F. Haughey, Project Manager
Project Directorate No. 3
Division of BWR Licensing

SUBJECT: NRR SALP INPUT - NINE MILE POINT NUCLEAR STATION UNIT 2

Enclosed is NRR input for the March, 1986 SALP Board meeting for Nine Mile Point Nuclear Station Unit 2. As discussed in the enclosure, our evaluation was conducted according to NRR Office Letter No. 44 dated January 3, 1984 and NRC manual chapter 0516, Systematic Assessment of Licensee Performance.

Mary Haughey
Mary Haughey, Project Manager
Project Directorate No. 3
Division of BWR Licensing

Enclosure:
As stated

Information in this record was deleted
in accordance with the Freedom of Information
Act, exemptions 5
FOIA- 91-17-2 (90-269) B-2

9102080332 12pp

K/3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket No. 50-410

FACILITY: Nine Mile Point Nuclear Station Unit 2

LICENSEE: Niagara Mohawk Power Corporation

EVALUATION PERIOD: February 1, 1985, to January 31, 1986

PROJECT MANAGER: Mary F. Haughey

I. INTRODUCTION

This report contains NRR's input to the SALP review for the Nine Mile Point Nuclear Station Unit 2 (NMP-2). The assessment of the licensee's performance was conducted according to NRR Office Letter No. 44, NRR Inputs to SALP Process, dated January 3, 1984. This Office Letter incorporates NRC Manual Chapter 0516, Systematic Assessment of Licensee Performance.

II. SUMMARY

NRC Manual Chapter 0516 specifies that each functional area evaluated will be assigned a performance category (Category 1, 2, or 3) based on a composite of a number of attributes. The performance of the Niagara Mohawk Power Corporation in the functional area of Licensing Activities is rated Category 2.

III. CRITERIA

The evaluation criteria used in this assessment are given in NRC Manual Chapter 0516 Appendix, Table 1, Evaluation Criteria with Attributes for Assessment of Licensee Performance.

IV. METHODOLOGY

This evaluation represents the integrated inputs of the Licensing Project Manager (LPM) and those technical reviewers who expended significant amounts of effort on NMP-2 licensing actions during the current rating period. Using the guidelines of NRC Manual Chapter 0516, the LPM, each reviewer and their middle management applied specific evaluation criteria to the relevant licensee performance attributes, as delineated in Chapter 0516, and assigned an overall rating category (1, 2, or 3) to each attribute. The reviewers included this information as part of each Safety Evaluation Report transmitted to the Division of Licensing. The LPM, after reviewing the inputs of the technical reviewers, combined this information with her own assessment of licensee performance and, using appropriate weighting factors, arrived at a composite rating for the applicant. A written evaluation was then prepared by the LPM and circulated to NRR management for comments. These comments were incorporated in the final draft.

The basis for this appraisal was the applicant's performance in support of licensing actions that were either completed or had a significant level of activity during the current rating period. These actions are as follows:

(1) Responses to the staff requests for information.

(2) Responses to outstanding and confirmatory issues in the SER.

(3) Presentations, responses and support for the ACRS full and subcommittee meetings.

(4) Support for NRR on-site audits during the SALP period.

(5) Response to the downcomer supports issue.

(6) Support of the Technical Specification review.

V. ASSESSMENT OF PERFORMANCE ATTRIBUTES

The applicant's performance evaluation is based on a consideration of five of the seven attributes specified in NRC Manual Chapter 0516. These are:

-- Management Involvement and Control in Assuring Quality

-- Approach to Resolution of Technical Issues from a Safety Standpoint

-- Responsiveness to NRC Initiatives

-- Staffing

-- Training

For the remaining two attributes (enforcement and reportable events), no basis exists for an NRR evaluation for the functional area of Licensing Activities.

Licensing Activities

1. Management Involvement and Control in Assuring Quality

In the past year there were two areas where management effort was intense for NMP-2: the ACRS full and subcommittee meetings and the downcomer support concern. [redacted]

A large number of audits were performed during this SALP period to support the licensing effort. [redacted] In Amendment 23 two statements appeared in Section 1 of the FSAR as follows:

> "The FSAR is not a design input document. Quantities, dimensions, and values may be stated as nominal; however, the variations from actual as-built/as-designed values are within the design acceptance criteria." (Section 1.1, page 1.1-3)

and

> "The drawings listed in this section are provided to assist the NRC in the FSAR review. For as-designed/as-built conditions, the latest controlled drawings are applicable." (Section 1.7, page 1.7-1)

[redacted] The FSAR states that these pumps are sized to inject 43 GPM boron solution per pump into the reactor. The SLCS is required to be capable of injecting 86 GPM of 13 weight percent solution of sodium pentaborate solution in accordance with 10 CFR 50.62 (and as discussed in SER Section 15.8). [redacted]

2. Approach to Resolution of Technical Issues from a Safety Standpoint

[redacted]

3. Responsiveness to NRC Initiatives

4. <u>Enforcement History</u>

No basis exists for an NRR evaluation for the functional area of Licensing Activities.

5. <u>Reporting and Analysis of Reportable Events</u>

No basis exists for an NRR evaluation for the functional area of Licensing Activities.

6. <u>Staffing</u>

[redacted]

7. Training

[redacted]

8. Conclusion

[redacted]

Other Review Areas (follows J. Linville memo 1/9/86)

1. Operations

A large number of procedures need to be completed. A number of SER confirmatory items cannot be closed until the associated procedures are completed and reviewed.

2. Training

See the same subject in the licensing area.

3. Radiological Controls

Additional testing is required of the isolators manufactured by Kaman. Completion of this testing may affect the completion of the Radiation Monitoring System. The applicant has not provided the NRC with a list of where these isolators are used.

4. Maintenance

[redacted]

5. PSI/ISI Program and Performance

[redacted]

6. Preoperational Testing

[redacted]

7. Fire Protection

[redacted]

8. Security

[redacted]

9. Construction

[redacted] (See Licensing item 3)

10. Quality Programs and Controls

No NRR input for this area.

Information to be Added to
Section V of the SALP Report - "Supporting Data and Summary"

1. NRR Licensee Meetings

A large number of meetings were held with the applicant in Bethesda to resolve/discuss staff concerns. These are documented by meeting summaries.

2. NRR Site Visits & Audits

Instrumentation and Control Audit	January 7, 8, & 9, 1986
Environmental Qualification Audit	December 16 - 20, 1985
Seismic Qualification Review Team Audit	July 8 - 12, 1986
Pump and Valve Operability Review Team Audit	July 8 - 12, 1986
Containment Systems Site Visit	January 7, 1986
Electrical Power Systems Site Visit	December 17 & 18, 1985
DCRDR Audit	March 19 - 22, 1985
SPDS Audit	July 17 & 18, 1985
Revetment Ditch Audit	August 27, 1985

3. Licensing Documents Issued

FES	April 1985
SER	February 1985
SSER-1	June 1985
SSER-2	November 1985
Draft Technical Specifications	August 29, 1985
Proof-and-Review Technical Specifications	November 20, 1985

4. Applicant Responses

   a. Responses to requests for information.

   b. Letters & FSAR updates to respond to SER concerns.

   c. Responses to ACRS questions.

   d. Responses to concerns on downcomer supports.

5. Support for the Technical Specification review.

6. Support for the ACRS full and subcommittee meetings.

HISTORY OF SALP RATINGS FOR THE PREVIOUS TWO RATING PERIODS

October 1982 - September 1983

SUMMARY OF RESULTS

NINE MILE POINT, UNIT 2

Functional Areas	Category 1	Category 2	Category 3
Soils and Foundations	X		
Containment and Other Safety Related Structures		X	
Piping Systems and Supports			X
Safety Related Components		X	
Support Systems	No basis for rating		
Electrical Power Supply and Distribution		X	
Instrumentation and Control Systems		X	
Licensing Activities		X	
Project Management/Quality Assurance			X

October 1983 - January 1985

Functional Area	Category Last Period (10-1-82 - 9-30-83)	Category This Period (10-1-83 - 1-31-85)	Recent Trend
Containment and other Safety Related Structures	2	2	Consistent
Piping Systems and Supports	3	2	Improving
Safety Related Components-Mechanical	2	1	Consistent
Support Systems	Not Assessed	1	Consistent
Electrical Equipment and Cables	2	3	Consistent
Instrumentation and Control Systems	2	2	Consistent
Licensing Activities	2	2	Consistent
Project Management/Quality Assurance	3	2	Improving
Nondestructive Examination	Not Assessed	2	Improving
Engineering	Not Assessed	3	Improving

NMP-2 SALP (Feb. 1985 - Jan. 1986)

MATRIX OF REVIEW BRANCH INPUTS

CRITERIA

Reviewer	Branch	Date	1	2	3	4	5	6	7
R. Wright	EQB	01/23/86	[redacted]						
D. Smith	MTEB	10/30/85							
F. Witt	CHEB	03/19/85							
R. Benedict	LQB	04/19/85							
B. Elliot	MTEB	07/29/85							
*F. Witt	CHEB	03/19/85							
J. Lane	CSB	05/05/85							
K. Desai	RSB	08/23/85							
A. Singh	ASB	09/05/85							
J. Read	AEB	08/13/85							
*J. Lane	CSB	09/13/85							
J. Mauck	ICSB	10/01/85							
Lomb./Romney	EQB	10/03/85							
M. Hum	MTEB	11/85							
S. Saba	HFEB	01/08/86							
E. Manili	NMSS	02/05/86							
J. Kudrick	CSB	02/86							
Average									

I = insufficient input
NA = not applicable
* = input from same reviewer not counted twice

CRITERIA

The following criteria were used as applicable in evaluation of each functional area:

1. Management involvement in assuring quality.

2. Approach to resolution of technical issues from a safety standpoint.

3. Responsiveness to NRC initiatives.

4. Enforcement history

5. Reporting and analysis of reportable events.

6. Staffing (including management).

7. Training effectiveness and qualification.

To provide consistent evaluation of licensee performance, attributes associated with each criterion and describing the characteristics applicable to Category 1 and 2 and 3 performance were applied as discussed in NRC Manual Chapter 0516, Part II and Table 1.