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See Rep'g

SUBJECT: Forwards marked-up Chapter 14 to FSAR, reflecting changes to accelerated test program per current commitments for review & approval. Justification & other changes startup test program also encl.

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

May 30, 1986
(NMP2L 0724)

Ms. Elinor G. Adensam, Director
BWR Project Directorate No. 3
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Washington, DC 20555

Dear Ms. Adensam:

Re: Nine Mile Point Unit 2
Docket No. 50-410

Niagara Mohawk plans to implement an accelerated power ascension test program for Nine Mile Point Unit 2. Certain aspects of the accelerated test program change current commitments in the Final Safety Analysis Report and require Nuclear Regulatory Commission approval. Attachment 1 provides marked-up changes to Chapter 14 showing our proposed program. Justification for these changes is provided in Attachment 2.

The Hope Creek accelerated program was approved in letters dated March 20, 1986, February 4, 1986 and January 22, 1986. Since the Unit 2 program is similar, we would appreciate your early review and approval.

In addition to the Chapter 14 changes related to the accelerated test program, we are also providing in Attachment 3 other changes to the Startup Test program. These changes are the result of a number of events including position title revision, Technical Specification development and testing results. We would appreciate an opportunity to discuss these changes at your earliest convenience.

Very truly yours,

CVMangan

C. V. Mangan
Senior Vice President

NLR:ja
1615G

Attachments

xc: M. Haughey
R. A. Gramm, NRC Resident Inspector
R. W. Starostecki
Project File (2)

B001
1/1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)

Niagara Mohawk Power Corporation) Docket No. 50-410

(Nine Mile Point Unit 2))

AFFIDAVIT

C. V. Mangan, being duly sworn, states that he is Senior Vice President of Niagara Mohawk Power Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission the documents attached hereto; and that all such documents are true and correct to the best of his knowledge, information and belief.

CVMangan

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 30th day of May, 1986.

Christine Austin
Notary Public in and for
Onondaga County, New York

My Commission expires:

CHRISTINE AUSTIN
Notary Public in the State of New York
Qualified in Onondaga Co. No. 4787687
My Commission Expires March 30, 1987

ATTACHMENT 1

TEST REDUCTION ITEMS

14.2.10.3 Power Testing from 25 to 100 Percent of Rated Output

The power test phase comprises the following tests, many of which are repeated several times at the different test levels. While a certain basic order of testing is maintained relative to power ascension, there is, nevertheless, considerable flexibility in the test sequence at a particular power level which may be used whenever it becomes operationally expedient. In no instance, however, is nuclear safety compromised.

1. Coolant chemistry tests and radiation surveys are made at various test levels to preserve a safe and efficient power increase.

2. Selected CRDs are scram-timed at various power levels to provide correlation with the initial data.

~~3. The effect of control rod movement on other parameters (e.g., electrical output, steam flow,~~
22 ~~and neutron flux level) is examined for different power conditions.~~

3. ~~4.~~ Following the first reasonably accurate APRM calibration (25 percent power), the IRMs are reset.

4. ~~5.~~ At test conditions 2, 3, and 6, the local power range monitors (LPRMs) are calibrated.

5. ~~6.~~ The APRMS are calibrated initially, ~~and~~ after major ~~at each new~~ power level changes, and following LPRM calibration.

6. ~~7.~~ Completion of the process computer checkout is made for all variables, and the various options are compared with independent calculations as soon as significant power levels are available.

7. ~~8.~~ Collection of data from the system expansion tests is completed for those piping systems that have not previously reached full operating temperatures.

8. ~~9.~~ The axial and radial power profiles are explored fully by means of the traversing incore probe (TIP) system at representative power levels during the power ascension.

9. ~~10.~~ Core performance evaluations are made at ~~all~~ test points above the 25-percent power level and for

selected flow conditions; the work involves determination of the core thermal power, maximum linear heat generation rate, minimum critical power ratio (MCPR), and other thermal parameters.

10. ~~11.~~ Overall plant stability in relation to minor perturbations is shown by the following group of tests that are made at all test points:

~~a. Core power-void mode response.~~

a. ~~b.~~ Pressure regulator set point change.

b. ~~c.~~ Water level setpoint change.

c. ~~d.~~ Bypass valve opening.

d. ~~e.~~ Recirculation flow setpoint change.

For the first of these tests, neutron flux (power) response on LPRM chambers is observed on control rod withdrawal. The next two tests require that the changes made approximate as closely as possible a step change in demand, while for the next test 22 the bypass valve is opened quickly. The remaining test is performed to properly adjust the control loop of the recirculation system. For all of these tests, plant performance is monitored by recording the transient behavior of numerous process variables, the one of principal interest being neutron flux. Other imposed transients are produced by step changes in demand core flow, simulating loss of a feedwater heater and failure of the operating pressure regulator to permit takeover by the backup regulator.

11. ~~12.~~ The category of major plant transients includes full closure of all MSIVs, fast closure of turbine generator control valves and stop valves, loss of the main generator and offsite power, tripping of a feedwater pump, and ~~[illegible]~~ trips of the recirculation pumps. The plant transient behavior is recorded for each test and the results may be compared with the acceptance criteria and the predicted design performance.

12. ~~13.~~ A test is made of the main steam safety relief valves in which leaktightness and general operability are demonstrated.

13. ~~14.~~ The jet pump flow instrumentation is calibrated at test conditions 3 and 6.

14. ~~15.~~ The as-built characteristics of the recirculation system are determined as soon as operating conditions permit full core flow.

14.2.11 Test Program Schedule

Preoperational and startup testing is planned to be conducted in accordance with the following schedule. This schedule is based on current information and is updated onsite to consider actual construction and testing progress. It is included to provide general information and sequence but is not considered to be identical to the schedules in use during the startup and test program.

1. The preoperational/acceptance test phase commences in December 1984 and continues until fuel loading in February 1986.

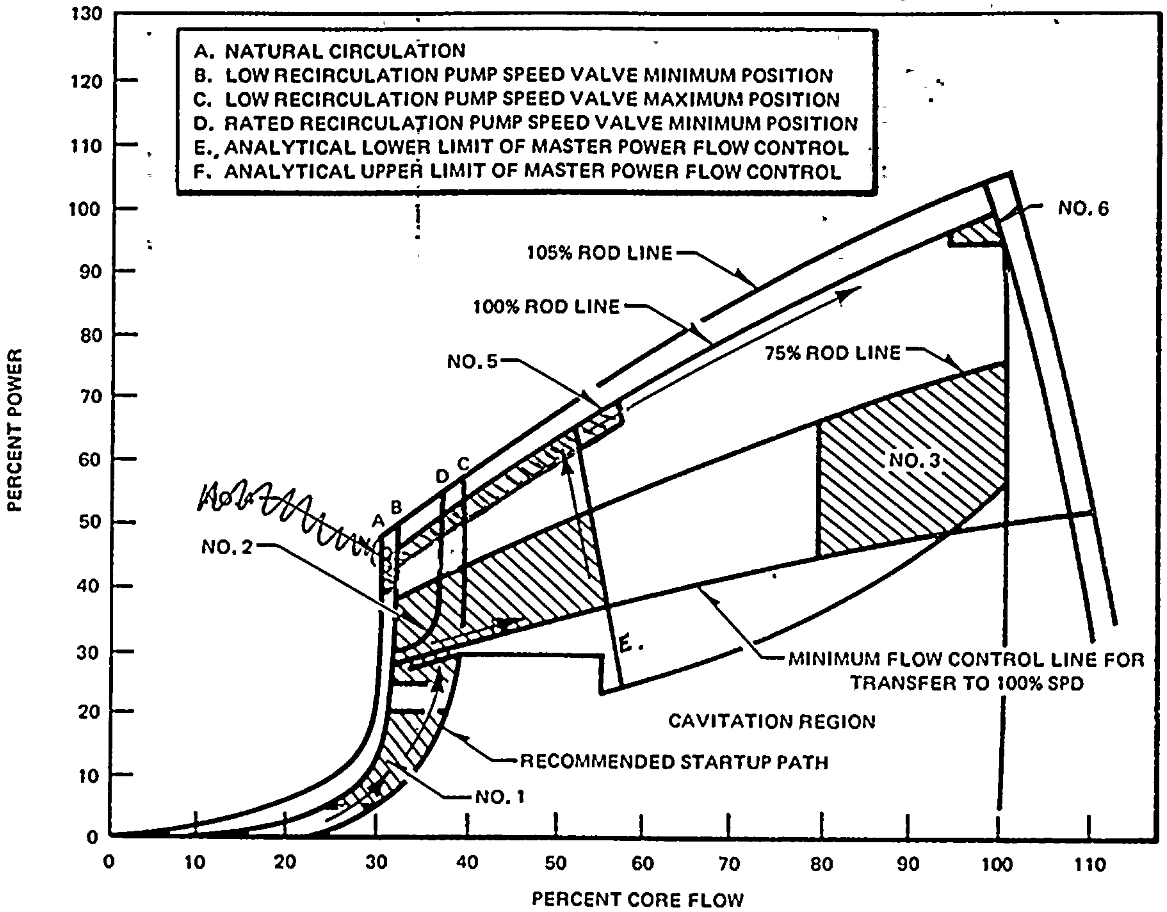
2. The startup test program commences with fuel load and continues through power ascension testing which is completed at the end of the 100-hr. warranty run in September 1986.

3. In general, approved preoperational test procedures will be available for NRC review at least 60 days prior to fuel load.

14.2.12 Individual Test Descriptions

14.2.12.1 Preoperational Tests

Test abstracts for the preoperational tests are provided in Tables 14.2-2 through 14.2-132. The abstracts identify each test by system; specify the major prerequisites and operating conditions necessary for each (mode of operations of major control systems); provide general test objectives, a summary of the test method, and a summary of the acceptance criteria. Some abstracts may require more than one test depending on variables such as plant status and availability, optimization of resources, and schedule restraints. When additional tests are required they are approved by the JTG, numbered and included on the current Test Index in accordance with the startup administrative procedures.



CONDITION (TC)

1 BEFORE MAIN GENERATOR SYNCHRONIZATION AND RECIRC PUMPS OPERATING ON LOW FREQUENCY POWER SUPPLY

2 BETWEEN 50% AND 75% CONTROL ROD LINES, AT OR BELOW THE ANALYTICAL LOWER LIMIT OF MASTER FLOW CONTROL MODE

3 FROM 50% TO 75% CONTROL ROD LINES AND CORE FLOW BETWEEN 80% AND MAXIMUM ALLOWABLE

5 From the 100% loadline to 5% below the 100% loadline and between minimum flow at rated recirculation pump speed (minimum valve position) to the natural circulation core flow line between the 100% rod line and the 80% rod line.

6 WITHIN 0 TO -5% OF RATED THERMAL POWER, AND WITHIN 5% OF RATED CORE FLOW RATE

~~AT 105% OF RATED THERMAL POWER AND 100% CORE FLOW~~

FIGURE 14.2-5

TEST CONDITION REGION DEFINITION

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
FINAL SAFETY ANALYSIS REPORT

TABLE 14.2-1 (Cont)

Table Number	System Number	Title
14.2-118	93	Rod Block Monitoring
14.2-119	94	TIP System
14.2-120	95A	Rod Worth Minimizer
14.2-121	95B	Rod Sequence Control
14.2-122	96	Reactor Manual Control and Control Rod Position Indication
14.2-123	97	Reactor Protection
14.2-124		DELETED
14.2-125	100A	Standby Diesel Generator
14.2-126	100B	HPCS Diesel Generator
24 14.2-127		DELETED
14.2-128	106	Redundant Reactivity Control System
14.2-129		Loss of Power/ECCS Functional
14.2-130		DELETED
14.2-131		Structural Integrity Integrated Leak Rate Test
14.2-132		Secondary Containment Leak Rate Test

C. Startup Tests

Table Number	Procedure Number	Title
14.2-201	SUT-1	Chemical and Radiochemical
14.2-202	SUT-2	Radiation Measurement
14.2-203	SUT-3	Fuel Loading
14.2-204	SUT-4B	Full Core Shutdown Margin *
14.2-205		DELETED
14.2-206	SUT-5	Control Rod Drive System
14.2-207	SUT-6	Source Range Monitor Performance
14.2-208	~~SUT-8~~	~~Control Rod Sequence Exchange~~ Deleted
14.2-209	SUT-10	Intermediate Range Monitor Performance
14.2-210	SUT-11	LPRM Calibration
14.2-211	SUT-12	APRM Calibration
14.2-212	SUT-13	~~NSSS~~ Process Computer *
14.2-213	SUT-14A	RCIC ~~System~~ *
14.2-214	SUT-16A	Selected Process Temperatures
14.2-215	SUT-16B	Water Level Reference Leg Temperature
14.2-216	SUT-17	System Expansion
14.2-217	SUT-18	TIP ~~Traversing Incore Probe~~ Uncertainty * ~~Calculations~~
14.2-218	SUT-19	Core Performance
14.2-219	SUT-20	Steam Production
14.2-220	~~SUT-21~~	~~Core Power-Void Mode~~ Deleted

TABLE 14.2-1 (Cont)

Table Number	Procedure Number	Title
14.2-221	SUT-22	Pressure Regulator
14.2-222	SUT-23A	Water Level Set Point, Manual Feedwater Flow Changes
14.2-223	SUT-23B	Loss of Feedwater Heating
14.2-224	SUT-23C	Feedwater Pump Trip
14.2-225	SUT-23D	Maximum Feedwater Runout Capability
14.2-226	SUT-24	Turbine Valve Surveillance
14.2-227	SUT-25A	Main Steam Isolation Valves Functional Tests
14.2-228	SUT-25B	Full Reactor Isolation
14.2-229		DELETED
14.2-230	SUT-26	Relief Valves
14.2-231	SUT-27	Turbine Trip and ~~Generator~~ Generator Load Rejection *
14.2-232	SUT-28	Same title as before (No Change) *
14.2-233	SUT-29A	Recirculation Flow Control, Valve Position Control
14.2-234	SUT-29B	Recirculation Flow Loop Control
14.2-235	SUT-30A	Recirculation System One Pump Trip
14.2-236	SUT-30B	Recirculation System Two Pump Trip
14.2-237	SUT-30C	Recirculation System Performance
14.2-238	SUT-30D	Recirculation Pump Runback
14.2-239	SUT-30E	Recirculation System Cavitation
14.2-240	SUT-31	Loss of Turbine Generator and Offsite Power ~~[illegible]~~
14.2-241	SUT-33	Drywell Piping Vibration *
14.2-242	SUT-35	Recirculation System Flow Calibration
14.2-243	SUT-70	Reactor Water Cleanup System
14.2-244	SUT-71	Residual Heat Removal System
14.2-245	SUT-74	Off-Gas System
14.2-301	SUT-75	Drywell Cooling System
14.2-302	SUT-76	ESF Area Cooling
14.2-303	SUT-77	BOP Piping Vibration
14.2-304	SUT-78 79	BOP System Expansion
14.2-305	SUT-~~79~~ 84	Reactor Internals Vibration
14.2-306	SUT-80	Emergency Recirculation Ventilation
14.2-401		DELETED
14.2-402		DELETED
14.2-403		Qualification of GE Principal Testing Personnel During Startup Testing
14.2-307	SUT-81	Drywell High Energy Penetrations *

22

TABLE 14.2-206

CONTROL ROD DRIVE SYSTEM

Startup Test (SUT-5)

Test Objectives

1. To demonstrate that the CRD system operates properly over the full range of primary coolant temperatures and pressures from ambient to operating.

2. To determine the initial operating characteristics of the entire CRD system.

Prerequisites

The appropriate preoperational tests have been completed. The SORC has reviewed and approved the test procedures and initiation of testing. The CRD manual control system preoperational testing must be completed on CRDs being tested. The reactor vessel, closed loop cooling water system, condensate supply system, and instrument air system must be operational to the extent required to conduct the test.

Test Procedure

The CRD tests performed during the startup test program are designed as an extension of the tests performed during the preoperational CRD system tests. Thus, after it is verified that all CRDs operate properly when installed, they are tested periodically during heatup to assure that there is no significant binding caused by thermal expansion of the core components. A list of all CRD tests to be performed during startup testing is as follows:

CONTROL ROD DRIVE SYSTEM TESTS

Action	Accumulator Pressure	Test Conditions Reactor Pressure with Core Loaded psig (kg/cm^2) 0	600(42.2)	800(56.2)	Rated
Position Indication		All			

12

TABLE 14.2-206 (Cont)

CONTROL ROD DRIVE SYSTEM TESTS

Action	Accumulator Pressure	Test Conditions Reactor Pressure With Core Loaded psig (kg/cm2) 0	600(42.2)	800(56.2)		Rated
Normal Stroke Times Insert/ Withdraw		All				4*
Coupling		All***				
Friction		All				4*
Scram	Normal	All	4*	4*	****	All
Scram	Minimum	4*				
Scram	Zero					4*
Scram	Normal					4**

* Refers to four CRDs selected for continuous monitoring based on slow normal accumulator pressure scram times or unusual operating characteristics, at zero reactor pressure or rated reactor pressure when this data is available. The four selected CRDs must be compatible with the rod worth minimizer, RSCS system, and CRD sequence requirements.

** Scram times of the four slowest CRDs (based on scram data at rated pressure) will be determined at test conditions 2 and 6 during planned reactor scrams.

*** Establish that this check is normal operating procedure.

NOTE: Single CRD scrams should be performed with the charging valve closed. (Do not ride the charging pump head.)

**** Individual scram times will be determined at test condition #2 in conjunction with the performance of two planned reactor scrams (see tests 28 and 31).

TABLE 14.2-218

CORE PERFORMANCE

Startup Test (SUT-19)

Test Objectives

1. To evaluate the core thermal power and flow.

2. To evaluate whether the following core performance parameters are within limits: 22

   a. MLHGR.

   b. MCPR.

   c. MAPLHGR.

Prerequisites

The preoperational tests have been completed and the SORC has reviewed and approved the test procedure and initiation of testing. System instrumentation has been installed and calibrated and test instrumentation calibrated.

Test Procedure

Core performance evaluation is employed to determine the principal thermal and hydraulic parameters associated with core behavior. These parameters are:

1. Core flow rate.

2. Core thermal power level.

3. MLHGR.

4. MAPLHGR.

5. MCPR.

These core performance parameters are evaluated by manual calculation techniques or may be obtained from the process computer. If the process computer is used as a primary means to obtain these parameters, it must be proven that it agrees with BUCLE within 2 percent on all thermal parameters (SUT-13). 22

TABLE 14.2-218 (Cont)

If neither BUCLE nor the process computer is available the manual calculation techniques can be used for the core performance evaluation.

The following test is performed:

Action	Test Conditions
Evaluate core thermal power, flow, and compute the thermal and hydraulic parameters associated with core behavior. Use plant process computer, offline computer system, or manual calculations	a. TC-1, 2, 3, ~~4~~, 5*, and 6 are necessary for documentation. b. Additional points as necessary to assure compliance with technical specifications.

Acceptance Criteria

Level 1:

1. The MLHGR of any rod during steady-state conditions does not exceed the limit specified by the plant technical specifications.

2. The steady-state MCPR does not exceed the limits specified by the technical specifications.

3. The MAPLHGR does not exceed the limits specified by the technical specifications.

22 4. Steady-state reactor power is limited to rated core thermal power and values on or below the rated power flow control line. Core flow does not exceed its rated value.

22 Level 2:

Not applicable.

* At mid power range and natural circulation

TABLE 14.2-220

CORE POWER-VOID MODE

Startup Test (SUT-21)

Test Objective

To measure the stability of the core power-void dynamic response and to demonstrate that its behavior is within specified limits.

Prerequisites

The appropriate preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and initiation of testing. System and test instrumentation have been installed and calibrated.

Test Procedure

The core power-void loop mode that results from a combination of the neutron kinetics and core thermal-hydraulic dynamics is least stable near the natural circulation end of the rated 100-percent power rod line. A fast change in the reactivity balance is obtained by a pressure regulator step change (see Test 22) and by moving a very high worth rod one or two notches. Both local flux and total core response will be evaluated by monitoring selected LPRMs during the transients.

The following test is performed:

Action	Test Conditions
1. Move high worth control rod 1 to 2 notches.	a. TC-4 natural circulation. 22 b. Low power region of TC-5 with recirculation flow control valve at minimum valve position. c. Low power region of TC-5 with LFMG power and minimum valve position. d. High power region of TC-5.
2. In conjunction with pressure regular step changes (Test 22).	a. TC-4 and TC-5. 22

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TABLE 14.2-220 (Cont)

Acceptance Criteria

Level 1:

12 The decay ratio of any oscillatory core variable must be less than 1.0 at all test points.

Level 2:

12 System-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.50.

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TABLE 14.2-221

PRESSURE REGULATOR

Startup Test (SUT-22)

Test Objectives

1. To determine the optimum settings for the pressure control loop by analysis of the transients induced in the reactor pressure control system by means of the pressure regulators.

2. To demonstrate the takeover capability of the backup pressure regulator upon failure of the controlling pressure regulator and to set spacing between the set points at an appropriate value.

3. To demonstrate smooth pressure control transition between the control valves and bypass valves when reactor steam generation exceeds steam used by the turbine.

4. To demonstrate that other affected parameters are within acceptable limits during pressure-regulator-induced transient maneuvers. 12

Prerequisites

The preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

The pressure set point is decreased rapidly and then increased rapidly by about 10 psi, and the response of the system is measured in each case. It is desirable to accomplish the set point change in less than 1 sec. At specified test conditions the load limit set point is set so that the transient is handled by control valves, bypass valves, or both. The backup regulator is tested by simulating a failure of the operating pressure regulator so that the backup regulator takes over control. The response of the system is measured and evaluated, and regulator settings are optimized. At certain conditions the test results will be included with the test report in core 12

TABLE 14.2-221 (Cont)

~~Power/Void Mode Response (Test 21) This testing yields~~ will provide valuable core stability data in the midfrequency range (i.e., 0.1 to 3.0 Hz).

Test No. 22 - Pressure Regulator

22

Action: Operating Mode	Input	Test Condition: 1	2	3	~~4~~	5	6
CV	Set point	No	Yes	Yes	~~Yes~~	Yes	Yes
CV	Fail to backup(3)	No	Yes	Yes	~~Yes~~	~~Yes~~ No	Yes
BPV	Set point	Yes	Yes	No	~~Yes~~	Yes	Yes
BPV	Fail to backup(3)	Yes	Yes	No	~~Yes~~	No	Yes
BPV-IN(2)	Set point	No	No	Yes	~~No~~	No	No
	Recirculation modes	MAN	MAN	MAN and FLX	~~MAN~~	MAN	MAN(1) and FLX

Acceptance Criteria

Level 1:

The transient response of any pressure control system-related variable to any test input must not diverge.

Level 2:

1. Pressure control system-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25. (This criterion does not apply to tests involving simulated failure of one regulator with the backup regulator taking over.)

(1) Either POS or FLO.
(2) Bypass valve incipient opening test, MAN mode only.
(3) Failure to bakcup regulator, MAN mode only.

TABLE 14.2-221 (Cont)

2. The pressure response time from initiation of pressure set point change to the turbine inlet pressure peak is ≤10 sec.

3. Pressure control system deadband, delay, etc, is small enough that steady-state limit cycles (if any) produce steam flow variations no larger than ±0.5 percent of rated steam flow.

4. For all pressure regulator transients the peak neutron flux and/or peak vessel pressure shall remain below the scram settings by 7.5 percent and 10 psi, respectively. (Maintain a plot of power versus the peak variable values along the 100-percent rod line.) 22

5. The variation in incremental regulation (ratio of the maximum to the minimum value of the quantity, "incremental change in pressure control signal/incremental change in steam flow," for each flow range) shall meet the following:

Steam Flow Obtained With Valves Wide Open (Percent)	Variation
0 to 85	≤4:1
85 to 97	≤2:1
85 to 99	≤5:1

TABLE 14.2-222

WATER LEVEL SET POINT,
MANUAL FEEDWATER FLOW CHANGES

Startup Test (SUT-23A)

Test Objective

To verify that the feedwater control system has been adjusted to provide acceptable reactor water level control.

Prerequisites

The preoperational tests have been completed; the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

Reactor water level set point changes of approximately 3 to 6 in are used to evaluate (and adjust if necessary) the feedwater control system settings for all power and feedwater flow control valve modes. The level set point changes will also demonstrate core stability to subcooling changes.

The following tests are performed:

STET
Yes

Action		Test Condition					
Operating Mode	Input	1	2	3	~~4~~	5	6
3-element	Set point	No	Yes	Yes	~~Yes~~	Yes ~~Yes~~	Yes
1-element	Set point	No	Yes	Yes	~~Yes~~	Yes	Yes
NORM	Manual flow steps*	No	Yes	Yes	~~No~~	No	Yes
	Recirculation modes**	MAN	MAN	MAN	~~MAN~~	MAN	MAN 12

*Manual flow steps to be done on each flow control valve only when two pumps or more are on, with one or more in automatic mode and the valve to be tested in manual mode.

**Either POS or FLO 12

Acceptance Criteria

Level 1

The transient response of any level control system-related variable to any test input must not diverge.

Level 2

1. Level control system-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25 in.

2. The open loop dynamic flow response of each feedwater actuator (control valve) to small (<10 percent NBR) step
22 disturbances shall be:

    a) Maximum time to 10 percent of a step disturbance — ≤~~1.1~~ 1.2 sec

    b) Maximum time from 10 percent to 90 percent of a step disturbance — ≤~~1.9~~ 2.1 sec

    c) Peak overshoot (percent of step disturbance) — ≤15 percent

    d) Settling time (100 percent
22 ±5 percent of step distribution) — ≤14.0 sec

3. The average rate of response of the feedwater actuator to large (>10 percent of NBR) step disturbances shall be
22 between 10 and 25 percent nuclear boiler rated feedwater flow/second. This average response rate will be assessed by determining the time required to pass linearly through the 10 percent and 90 percent response points.

4. At steady-state operation for the 3/1 element systems, input scaling to the mismatch gains should be adjusted
22 such that the level error due to biased mismatch gain output should be within ±1 in.

TABLE 14.2-223 (Cont)

Level 2:

22 The increase in simulated heat flux does not exceed the predicted value referenced to the actual feedwater temperature change and initial power level.

TABLE 14.2-230

RELIEF VALVES

Startup Test (SUT-26)

Test Objectives

1. To verify that the relief valves function properly (can be opened and closed manually).

2. To verify that the relief valves reseat properly after operation.

3. To verify that there are no major blockages in the relief valve discharge piping.

Prerequisites

The preoperational tests have been completed, the SORC has reviewed and approved the test procedures and initiation of testing, and instrumentation has been checked or calibrated as appropriate.

Test Procedure

with steamflow greater than the individual relief valve capacity

A functional test of each SRV is made as early in the startup program as practical. This is normally the first time the plant reaches ~~250~~ 950 psig. ~~The test is then repeated at rated reactor pressure. Bypass valve (BPV) response is monitored during the low-pressure test and the~~ electrical Bypass valve or output response is monitored during the ~~rated pressure~~ test. The test duration is about 10 sec to allow turbine valves and tailpipe sensors to reach a steady state.

The tailpipe sensor responses are used to detect BPV or the opening and subsequent closure of each SRV. The ~~BPV and~~ power level (MWe) response~~s~~ ~~are~~ is analyzed for anomalies indicating a restriction in an SRV tailpipe. In addition, lead BWR plants measure SRV tailpipe back pressure on the longest and shortest tailpipes.

Valve capacity is based on certification by ASME code stamp and the applicable documentation being available in the onsite records. The nameplate capacity/pressure rating assumes that the flow is sonic. This is true if the back pressure is not excessive. A minor blockage of the line may prevent sonic flow and it should be determined that no major 22

TABLE 14.2-230 (Cont)

22 blockage exists through the BPV or MWe response signatures.

The following tests are performed:

Action	Test Conditions
~~1. STS: 10-sec manual opening for functional check of valve and sensor response.~~	~~a. Heatup at 250 psig.~~ ~~b. STS: Recirculation system in manual mode. Other systems in NORM mode.~~
1. 2. Manual opening for plant response and ~~valve~~ reseating checks. ~~SRV is opened.~~ SUT-~~33~~ 77, Drywell Piping vibration, is to be done in conjunction with this test.	a. At ≥ 950 psig. (TC-1) ~~Between TC-2 and 3~~, if any valve is readjusted, repeat test. b. Recirculation system in MAN mode. Other systems in NORM mode.

Acceptance Criteria

Level 1:

There is positive indication of steam discharge during the manual actuation of each valve.

Level 2:

1. Pressure control system-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response is less than or equal to 0.25.

TABLE 14.2-230 (Cont)

2. The temperature measured by thermocouples on the discharge side of the valves returns to within 10°F of the temperature recorded before the valve was opened. If pressure sensors are available, they return to their initial states upon valve closure.

~~3. During the 250-psig functional test the steam flow through each relief valve shall not be less than 90 percent of the average relief valve steam flow, as measured by bypass valve position.~~ 22

3. ~~4. During the rated pressure test the steam flow through each relief valve, as measured by MWe, is not less than 0.5 percent of rated MWe less than the average of all the valve responses.~~

3. During the test the steam flow through each relief valve shall not be less than 90% of the average relief valve steam flow as measured by bypass valve position or the steam flow through each relief valve as measured by MWe shall not be lower than the average valve response by more than 0.5% of rated MWe.

TABLE 14.2-231

TURBINE TRIP AND GENERATOR LOAD REJECTION

Startup Test (SUT-27)

Test Objective

To demonstrate the response of the reactor and its control systems to protective trips in the turbine and generator.

Prerequisites

The appropriate preoperational tests have been completed; the SORC has reviewed and approved the test procedures and initiation of testing. All controls and interlocks are checked and instrumentation calibrated.

Test Procedure

1. Turbine trip (closure of the main turbine stop valves within 0.1 sec) and generator trip (closure of the main turbine control valves within 0.3 sec) is performed at selected power levels during the startup test program. At low power levels, reactor protection following the trip is provided by high neutron flux and vessel high-pressure scrams. For the protective trips occurring at intermediate and higher power levels, the reactor scrams by relays, actuated by stop/control valve motion.

2. A turbine ~~generator~~ trip is performed at low power level in such a way that nuclear boiler steam generation is just within the bypass valve capacity to demonstrate scram avoidance.

~~3. For the trips performed at intermediate power range, reactor scram is most important in controlling the transient peaks.~~

3, ~~4.~~ Above 30-percent power, the recirculation pump circuit breakers are both automatically tripped and subsequent transient pressure rise is limited by the opening of the bypass valves initially, and the safety relief valves, if necessary.

4. ~~5.~~ For the turbine trip, the main generator breakers remain loaded for a time so there is no rise in turbine generator speed, whereas in the generator trip, the main

TABLE 14.2-231 (Cont)

generator breaker opens and the residual turbine steam causes a momentary rise in the generator speed.

The following tests are performed:

Action	Test Conditions
12 1. Main Turbine ~~Generator load rejection~~, ~~main breaker~~ trip. 12 ~~(SUT-33, Drywell Piping Vibration, is to be done in conjunction with this test.)~~ (SUT 77, BOP Piping Vibration is to be done in conjunction with ~~the~~ this test).	a. At TC-1 or 2, just within bypass system capacity. b. Recirculation system in FLO mode; other systems in NORM mode. c. Manual intervention permissible to prevent high or low water level trip.
~~2. Main turbine trip scram. (SUT-33, Drywell Piping Vibration, is to be done in conjunction with this test.)~~	~~a. TC-3 (60-80% power) at ≥95% core flow. b. All systems in NORM mode.~~
12 ~~3.~~ 2. Generator trip scram. (SUT-33, Drywell Piping Vibration, and SUT 77 are ~~is~~ to be done in conjunction with this test.)	a. Will be done at TC-6. b. All systems in NORM mode.

Previous experience demonstrates that reactor responses to a turbine trip and a generator load rejection at full power are similar for plants like Unit 2 which have steam bypass capacities equivalent to approximately 25 percent of rated power. The load rejection trip is performed at full power to test the turbine overspeed protection system.

Acceptance Criteria

Level 1:

12 1. For turbine and generator trips at power levels greater than 50 percent NBR, there is a delay of less than 0.1 sec following the beginning of control or stop valve closure before the beginning of bypass valve opening. The bypass valves are opened to a point corresponding to greater than or equal to 80 percent of their capacity

TABLE 14.2-234 (Cont)

Scram Avoidance and General Criteria

Level 1:

Not applicable.

Level 2:

For any one of the above loops' test maneuvers, the trip avoidance margins must be at least the following:

1. For APRM ≥7.5 percent.

2. For simulated heat flux ≥5.0 percent.

~~3. The load following loop response must produce steam flow variations no larger than 0.5 percent of rated steam flow.~~

TABLE 14.2-237

RECIRCULATION SYSTEM PERFORMANCE

Startup Test (SUT-30C)

Test Objective

To record recirculation system parameters during the power test program.

Prerequisites

The preoperational tests are complete. The SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

Recirculation system parameters are recorded at several power-flow conditions and in conjunction with single pump trip recoveries.

The following test is performed:

Action	Test Conditions
Record steady-state operating data.	a. At TC-2, 3, ~~4~~ 5*, and 6. b. During recovery from single pump trips of SUT-30A.

Acceptance Criteria

Level 1:

Not applicable.

Level ~~2~~ 3:

1. The core flow shortfall shall not exceed 5 percent at rated power.

2. The measured core plate ΔP shall not be >0.6 psi above prediction.

~~3. The calculated jet pump M ratio shall not be <0.2 points below prediction.~~

22

Level 2:

1. The measured core plate ΔP shall ~~not~~ be ≥3.0 psi above prediction.

* At natural circulation

TABLE 14.2-237 (Cont)

3 ~~4~~. The drive flow shortfall shall not exceed 5 percent at rated power.

4 ~~5~~. The measured recirculation pump efficiency shall not be >8 percent points below the vendor-tested efficiency.

~~6. The nozzle and riser plugging criteria shall not be exceeded.~~

TABLE 14.2-239

RECIRCULATION SYSTEM CAVITATION

Startup Test (SUT-30E)

Test Objective

To verify that no recirculation system cavitation occurs in the operable region of the power-flow map.

Prerequisites

The appropriate preoperational tests have been completed. The SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

Both the jet pumps and the recirculation pumps will cavitate at conditions of high flow and low power where NPSH demands are high and little feedwater subcooling occurs. However, the recirculation flow will automatically run back upon sensing a decrease in subcooling (as measured by the difference between the steam dome and recirculation loop temperature) to lower the reactor power. It will be verified that these limits are sufficient to prevent operation where recirculation pump or jet pump cavitation is predicted to occur. 22

The recirculation system flow control valves will cavitate at conditions of high differential pressure and low power (low subcooling). The recirculation flow will automatically run back upon sensing a decrease in subcooling (as measured by a low feedwater flow). This limit will be verified to ensure that operation is prevented where flow control valve cavitation may occur.

In both of these cases, flow runback is accomplished by a shift in the power supply to the recirculation pump motors from normal power to the low frequency motor generators. However, actual transfer to low frequency may not be required during this test as long as no sign of recirculation pump, jet pump, or flow control valve cavitation is evidenced.

TABLE 14.2-239 (Cont)

The following test is performed:

Action	Test Conditions
12 Insert control rods until cavitation occurs or until a cavitation interlock signal initiates ~~recirculation pump runback~~, whichever occurs first.	a. At TC-2 and 3. b. All systems in NORM mode.

Acceptance Criteria

Level 1:

Not applicable.

Level 2:

Runback logic settings are adequate to prevent operation in areas of potential cavitation.

Attachment 2 - Technical Evaluations

ITEM	TITLE		DESCRIPTION
5.1	ST-5	CRD/Hot Friction Testing	Test Simplification - Reduced Number of Tests
5.2	ST-5	CRD/Hot Single Rod	In Conjunction with Test 28
8.0	ST-8	Control Rod Sequence Exchange	Defer Testing Until Post-Commercial Operation
21.0	ST-21	Core Power/Void Mode	Justify Test Deletion
22.0	ST-22	Pressure Regulator	Test Simplification - Reduced Number of Tests
23.1	ST-23A	Feedwater Response	Test Simplification - Reduced Number of Tests
26.0	ST-26	Relief Valves	Test Simplification - Reduced Number of Tests
27.0	ST-27	Turbine Trip and Load Rejection	Test Simplification - Reduced Number of Tests
30.5	ST-30E	Recirculation System Cavitation	Test Simplification
71.0	ST-71	RHR System	Defer Non-Essential Equipment Testing Shutdown Cooling Mode
Test Condition 4			Natural Circulation Operation Test Simplification - Delete Test Condition

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 2.b requires friction testing (for BWR's) of control rods after the core is fully loaded. Startup Test 5, Control Rod Drive (CRD) System, performs friction testing of all CRD's at both rated pressure/temperature and cold conditions. It is proposed to reduce the number of CRD's to be hot friction tested to four.

DISCUSSION:

Performance of the control rod drives during friction testing is compared to acceptance criteria which require that during continuous insertion, the differential pressure variation for the CRD must not exceed a specified limit. If the limit is exceeded during continuous insertion, a settling test is performed to determine the differential settling pressure and variation.

CRD friction testing at the cold condition satisfies the objectives of Regulatory Guide 1.68, Appendix A, paragraph 2.b. The additional testing of Startup Test 5 at rated pressure/temperature is not required by the regulations and is often impractical to perform at these conditions. The hot condition represents a safety hazard to the technicians that operate the instrumentation required for friction testing. Substantial testing of CRD systems at previous BWR's has shown that testing of four selected CRD's at rated pressure conditions provides adequate information on the response of the system since all of the CRD's will be scram tested at rated pressure conditions. Testing at previous BWR/5 plants (LaSalle-1 and 2 and Hanford-2) has been performed on only four CRD's at rated pressure. In addition, more recent testing of BWR/6 plants has performed friction testing on all drives at rated pressure and none have failed the criteria.

Therefore, it is unneccessary to friction test all the CRD's at hot conditions and it is recommended that only four CRD's be hot friction tested to assure that there is no widespread problem because of thermal expansion. The four CRD's chosen should be the same as those for which scram timing is to be repeated during plant heatup.

CONCLUSION:

CRD friction testing at the cold condition during Startup Test 5 demonstrates the acceptable performance of the CRD system and satisfies the objectives of Regulatory Guide 1.68, Appendix A, paragraph 2.b for friction testing. The proposed change to reduce the number of CRD's to be hot friction tested does not adversely affect any safety systems or safe operation of the plant and therefore does not involve an unreviewed safety question. Startup Test 5, Control Rod Drive System, can therefore be simplified by reducing the number of CRD's to be hot friction tested to four.

STARTUP TEST 5 - CONTROL ROD DRIVE/HOT SINGLE ROD SCRAM TESTING
IN CONJUNCTION WITH TEST NUMBERS 28 AND 31

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 2.b, 4.o and 5.h require that control rod scram testing be performed at plant conditions that bound those under which the control rods might be required to function to achieve plant shutdown. The control rods are required to scram within the times specified by Plant Technical Specifications and assumed in FSAR analysis. Startup Test 5, Control Rod Drive System, demonstrates the performance of the control rod drive (CRD) system over the full range of reactor operating conditions. It is proposed to replace scram time testing of all but four CRD's at rated reactor pressure during heatup with the rated reactor pressure scram data from two planned full core scrams (in different control rod sequences) at Test Condition 2 (approximately 25% power).

DISCUSSION:

Scram performance of the control rod drive system is compared to acceptance criteria which require that the control rods scram within a specified time. Control rod drive scram timing tests are begun during preoperational testing and all later testing is an extension of the preoperational program. Following fuel loading, with the reactor in the cold shutdown condition, all CRD's are individually withdrawn and scram time tested. This test provides a data base from which four control rods are selected with scram times among the longest of those measured or which exhibit unusual operating characteristics. These four CRD's are individually scram time tested during heatup at approximately 600 and 800 psig reactor pressure.

Currently, at rated reactor pressure and low power level, all of the CRD's are again scram time tested. From this data base, four CRD's are selected for additional individual testing and for monitoring during full reactor scrams during the power ascension test program. It is proposed to only test four rather than all of the CRD's at rated reactor pressure during heatup. Scram times for the remaining CRD's would be obtained from the planned reactor scrams during performance of the

Shutdown from Outside the Control Room (Startup Test 28) and Loss of Turbine Generator and Offsite Power (Startup Test 31) at Test Condition 2 (approximately 25% power). These two full reactor scrams would provide scram times for all CRD's. This is possible because the two scrams would be performed with the reactor operating in different control rod sequences. Four or more CRD's would be selected for monitoring during future full reactor scrams.

Deferment of testing for all the CRD's from rated reactor pressure/low power to about 25% power is considered acceptable because (1) all CRD's are scram time tested at cold conditions, and (2) four CRD's are monitored during reactor heatup to rated pressure to determine the effects of increasing temperature and pressure. The combination of this data provides an indication of the performance of all CRD's at rated temperature and pressure. Increasing reactor power level should have little effect on CRD scram times. However scram times for all withdrawn control rods would be measured during the first two planned, full reactor scrams.

CONCLUSION:

Testing of the control rod scram times at hot conditions in conjunction with Startup Test 28, Shutdown from Outside the Control Room and Startup Test 31, Loss of Turbine Generator and Offsite Power at Test Condition 2 demonstrates the acceptable performance of the control rod drive scram system at hot reactor pressures. This proposed testing satisfies the objectives of Regulatory Guide 1.68, Appendix A, paragraphs 2.b, 4.o and 5.h, as well as satisfying the Technical Specification requirements for control rod scram times and therefore will not adversely affect any safety systems or safe operation of the plant and as such does not involve an unreviewed safety question. Therefore, Startup Test 5, Control Rod Drive hot scram time testing can be performed in conjunction with Startup Tests 28 and 31 at Test Condition 2.

STARTUP TEST 8 CONTROL ROD SEQUENCE EXCHANGE DEFER TESTING UNTIL FIRST REGULARLY SCHEDULED SEQUENCE EXCHANGE AT POWER DURING THE FIRST FUEL CYCLE

Objective:

Regulatory Guide 1.68 (Revision 2; August 1978), Appendix A, paragraph 5.c. requires a demonstration that core limits will not be exceeded during or following a control rod sequence exchange. It is proposed to defer performance of a sequence exchange until the first regularly scheduled exchange during the first fuel cycle.

Discussion:

Control rod sequence exchanges at Nine Mile Point Unit #2 will be performed in accordance with approved station procedures. These procedures implement the PCIOMR guidelines which are established to maintain acceptable fuel nodal power levels during a sequence exchange. The PCIOMR guidelines are recommendations designed to minimize plant capacity factor losses and optimize BWR core performance. Niagara Mohawk Power Corporation operations personnel have successfully implemented this approach at Nine Mile Point Unit #1. The main area of concern regarding a sequence exchange is to prevent fuel linear heat generation rate (LHGR) from exceeding its precondition envelope or threshold value as well as its Technical Specification limit. Performance of a sequence exchange during the startup test program will not provide meaningful data because the high enrichment bundles are barrier bundles and the medium enrichment bundles will not have received sufficient exposure during the test program to experience relatively low threshold values.

Conclusion:

Compliance with the PCIOMR guidelines is not a regulatory requirement; however, NMPC has demonstrated implementation of these guidelines at Nine Mile Point Unit #1. Performance of a sequence exchange during the test program will not provide as meaningful data as could be obtained later in the fuel cycle. Therefore, deferring performance of Startup Test 8 until the first regularly scheduled sequence exchange during the first fuel cycle does not adversely affect any safety systems or the safe operation of the plant and therefore does not involve an unreviewed safety question.

STARTUP TEST 21 - CORE POWER-VOID MODE RESPONSE
JUSTIFY TEST DELETION

OBJECTIVE:

There are no specific Regulatory Guide 1.68 requirements to perform stability testing during the power ascension program. However, paragraphs 5.s, 5.v and 5.h.h require the demonstration of acceptable control system responses during steady state and transient conditions. Startup Test 21, Core Power-Void Mode Response, measures the stability of the core power-void dynamic response by moving a very high worth control rod one or two notches. In conjunction, Startup Test 22, Pressure Regulator, performs pressure regulator step changes to measure the core power-void dynamic response. These tests are currently planned to be performed at Test Conditions 4 and 5. It is proposed to delete the control rod movement tests at Test Conditions 4 and 5.

DISCUSSION:

Acceptable response of the core power-void mode is determined by analyzing test data and comparing to an acceptance criterion which defines the required system performance. The criterion requires that all system related variables must exhibit non-divergent behavior. System related variables are heat flux and reactor pressure.

Measurement of system stability by movement of control rods was developed for small reactor cores. Use of this technique for large loosely coupled BWRs, typical of current plants in the startup testing phase, will not provide significant information on the stability of the system because of the low signal-to-noise ratio. In addition, for large BWR cores (i.e., Nine Mile Point - Unit 2), control rod worths in the power/flow range of interest are much less than for a small tightly coupled core. Instead, core wide disturbances provide more meaningful data for large cores. Startup Test 22, Pressure Regulator testing, measures the system reponse to pressure disturbances caused by actions

of the pressure regulator system. This testing yields valuable core stability data at the limiting high power/low flow condition encountered on the power/flow map (Test Condition 4). In addition to Startup Test 22, normal observations of operational power maneuvers provide sufficient data to determine the normal stability characteristics and response of the system.

In addition to the pressure regulator testing, Service Information Letter (SIL) 380 (Reference 1) provides detailed recommendations for the monitoring of system behavior. These recommendations provide for monitoring of neutron flux characteristics during normal operation at high power/low flow conditions and during abnormal operating conditions. In addition to the monitoring requirements, current Technical Specifications do not allow continued operation at natural circulation flow which is the least stable condition of the operating region.

Extensive special testing of stability characteristics has also been performed at several BWR's, including Vermont Yankee, Caorso, Leibstadt, Peach Bottom-2 and Browns Ferry. The test data has demonstrated the stability characteristics of BWR's over a wide range of conditions and has been reviewed along with extensive supporting analyses, as part of the Staff's Safety Evaluation Report on core thermal-hydraulic stability (Reference 2).

CONCLUSION:

As a result of the extensive testing and analysis of core thermal hydraulic stability, it has been demonstrated that General Electric BWR fuel and core designs meet the stability criteria set forth in General Design Criteria 10 and 12 of 10CFR50, Appendix A (Reference 2). Based on the above discussion and the Staff's Safety Evaluation Report (Reference 2), the proposed change will not adversely affect any safety related systems or safe operation of the plant and therefore does

not involve an unreviewed safety question. System stability is adequately measured during Startup Test 22, Pressure Regulator, and has been extensively tested at several BWRs covering a wide range of designs. In addition, information on the system's stability is continuously provided by SIL-380 recommendations for the monitoring of neutron flux. Therefore, Startup Test 21, Core Power-Void Mode Response can be deleted from the Power Ascension Test Program.

References:

1. "BWR Core Thermal Hydraulic Stability", Service Information Letter 380, Revision 1, General Electric Company, February 10, 1984.

2. Letter, C.O. Thomas (NRC) to H.C. Pfefferlen (GE), "Acceptance For Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, 'Thermal Hydraulic Stability Amendment to GESTAR II'", April 24, 1985.

STARTUP TEST 22 - PRESSURE REGULATOR
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2; August 1978), Appendix A, paragraphs 4.u and 5.s require demonstration of the operability during low power testing and calibration and performance verification during power ascension testing of the Pressure Regulator system. Startup Test 22, Pressure Regulator, determines the response of the system to rapid pressure setpoint changes and at specified test conditions the load limit setpoint will be set so that the transient is handled by control and/or bypass valves. Backup pressure regulator takeover will be tested by simulating a failure of the selected pressure regulator. Performance of this test is planned for Test Conditions 1-6. It is proposed to delete the Pressure Regulator tests at Test Condition 4, the backup pressure regulator takeover testing at Test Condition 5, and the Automatic Load Following mode tests.

DISCUSSION:

The Automatic Load Following (ALF) mode of operation is not a safety related function, but, is instead an operational improvement. Since it consists of non-essential equipment, the ALF mode may be disabled and no testing of the ALF mode is required for the Pressure Regulator System.

The pressure regulator system is primarily sensitive to vessel steam flow (and hence, power level) since the reactor is basically operated as a constant pressure device for varying steam flows. Therefore, testing of the pressure regulator response should cover the range of expected core power levels and is not significantly dependent on core flow since the steam flow at a fixed power level is insensitive to the core flow rate. Testing of the pressure regulator system during Test Condition 2, 3, 5 and 6 adequately covers the range of expected power levels during plant operation. Therefore, testing of the pressure regulator system at Test Condition 4 is not required for verification of the controller performance, and testing at Test Conditions 2, 3, 5 and 6 will provide

adequate confirmation of the system performance over the entire operating range.

Pressure regulator testing (specifically the pressure setpoint changes) at Test Condition 4 also provides information on the stability of the system. However, information on the stability of the reactor at natural circulation can also be obtained by monitoring the neutron flux (both local and core average) as recommended by the Service Information Letter (SIL) Number 380, Revision 1 (Reference 1). These surveillance recommendations provide for monitoring of the APRM and LPRM detectors when operating at natural circulation conditions and provide sufficient information on the stability of the reactor at natural circulation in addition to providing operator training for the monitoring procedures. Pressure regulator testing (pressure setpoint changes) will be performed at Test Condition 5, which bounds the least stable portion of the normal operating region, and will provide additional information on the stability of the reactor. Therefore, pressure regulator testing at Test Condition 4 can be deleted.

Testing of the backup pressure regulator is performed by simulating the failure of a selected pressure regulator. This test is currently planned to be performed at Test Conditions 2, 3, 5 and 6. Testing at Test Conditions 2, 3 and 6 provides adequate demonstration of the capability of the backup pressure regulator to control pressure in the event of a failure of the controlling pressure regulator since these test conditions bound the power level of Test Condition 5. Therefore, testing of the backup pressure regulator at Test Condition 5 can be deleted.

CONCLUSION:

Disabling of the ALF mode of operation does not adversely affect any safety related systems or the safe operation of the plant and does not involve an unreviewed safety question. Testing of the Pressure Regulator system at Test Condition 4 is not required since the pressure regulator will be tested at other Test Conditions that bound the power level of Test Condition 4. Stability data will be obtained during natural circulation by monitoring of the APRM and LPRM detectors as recommended by SIL-380, Revision 1. Therefore, deletion of Test Condition 4 testing does not adversely affect any safety related systems or the safe operation of the plant and does not involve an unreviewed safety question. Testing of the backup pressure regulator at Test Conditions 2, 3 and 6 demonstrates the performance of the backup system. Deleting testing of the backup pressure regulator at Test Condition 5 does not adversely affect any safety systems or the safe operation of the plant and therefore does not involve an unreviewed safety question. Startup Test 22, Pressure Regulator, can therefore be simplified by deleting the ALF testing at Test Condition 5, backup pressure regulator testing at Test Condition 5, and all pressure regulator testing at Test Condition 4.

REFERENCES:

1. "BWR Core Thermal Hydraulic Stability," Service Information Letter 380, Revision 1, General Electric Company, February 10,1984.

STARTUP TEST 23A - FEEDWATER SYSTEM RESPONSE
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 4.u, 5.s and 5.v require the demonstration of operability during low power testing, the calibration and performance verification during power ascension testing and the verification of operation in accordance with design requirements during power ascension testing of the feedwater control system. Startup Test 23A, Feedwater System Response, verifies that the feedwater system has been adjusted to provide acceptable reactor water level control. Nominal water level setpoint changes are used to evaluate the feedwater control system settings for all power and feedwater pump modes. Performance of this test is currently planned for Test Conditions 2-6. It is proposed to delete water level setpoint and manual feedwater flow tests at Test Condition 4, and to relax open loop dynamic valve position response acceptance criteria.

DISCUSSION:

Acceptance criteria define acceptable performance of the feedwater control system to testing perturbations. Criteria require that the transient response of any level control system-related variable to any test input must be non-divergent and also specify response characteristics for given disturbances. Testing during the power ascension program is performed at Test Conditions 2-6 to demonstrate compliance to these criteria.

The feedwater control system maintains the mass balance of the reactor vessel by supplying water to the vessel to match the steam flow exiting the vessel, thereby maintaining a constant water level during normal operation. Therefore, the feedwater control system is primarily dependent on the vessel steam flow and hence the reactor power. Testing of the feedwater control system at Test Conditions 2, 3, 5 and 6 adequately bounds the expected power levels for system operation. Since the power level of Test Condition 4 is similar to that of Test Condition 5, the feedwater control system performance at Test Condition 4 is not expected to be significantly different than at Test Condition 5. Therefore, testing at Test Conditions

2, 3, 5 and 6 will adequately confirm the system performance over the entire operating range. The objectives of Regulatory Guide 1.68, paragraphs 4.u, 5.s and 5.v are still satisfied with the remaining testing.

The feedwater control system response to small disturbances is measured during testing and compared to acceptance criteria which generally govern the system response time and damping of the step disturbances. The required criteria provide conservative limits for the system response to assure that the response assumed for safety analysis is met during actual plant operation. For optimum tuning of the Feedwater Control System, some of the criteria may be relaxed to provide more realistic limits for system response. For the open loop dynamic valve position response of each flow control valve, the acceptance criteria for the maximum time to 10% of a step disturbance and the maximum time from 10% to 90% of a step disturbance can be relaxed by 10%. Demonstration of compliance to these relaxed criteria will still ensure that the plant response is consistent with that assumed in the safety analysis. Therefore, these changes do not affect a safety system or the safe operation of the plant and as such do not result in an unreviewed safety question.

CONCLUSION:

Testing of the feedwater control system at Test Conditions 2, 3, 5 and 6 provides demonstration of system performance over the entire operating range. As such, deletion of Test Condition 4 testing does not adversely affect any safety systems or the safe operation of the plant and as such does not involve an unreviewed safety question. The proposed testing satisfies the objectives of Regulatory Guide 1.68, Appendix A, paragraphs 4.u, 5.s and 5.v, as well as the requirements of Startup Test 23A. Therefore, Startup Test 23A, Feedwater System Response, can be simplified by deleting testing at Test Condition 4, and relaxing Level 2 testing criteria.

STARTUP TEST 26 - RELIEF VALVES
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 4.p and 5.t require that the operability, response times, relieving capacities, setpoints and reset pressures for main steam line relief valves be verified. Startup Test 26, Relief Valves, demonstrates the proper operation of the main steam relief valves. Operability testing is currently planned to be performed during heatup at low pressure (250 psig) and between Test Condition 2 and 3. In addition, Plant Technical Specifications require surveillance testing of the Automatic Depressurization System (ADS) valves. It is proposed to replace the startup testing during heatup and between Test Condition 2 and 3 with testing at Test Condition 1.

DISCUSSION:

Acceptable response of the relief valves is determined by analyzing test data and comparing to acceptance criteria which define the required system performance. The criteria require that there is positive indication of steam discharge during a manual actuation of each valve, that the pressure control system response is stable, and that the discharge temperature remains within acceptable limits. These criteria can be demonstrated during single valve testing at rated reactor pressure with steam flow greater than the relief valve capacity.

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 4.p requires demonstration of relief valve operability at rated temperature during low power testing (defined as "normally at less than 5% power"). To provide adequate control of system pressure, the

testing must be performed at a steam flow that is greater than the individual relief valve capacity. Since the relief valve capacity is typically 5-7% steam flow, testing is proposed to be done at approximately 10-20% power (Test Condition 1), approximately the test power level of paragraph 5.t of Regulatory Guide 1.68.

Actuation of relief valves at low pressure has been identified as a contributor to valve seat damage caused by reseating against abnormally low pressure. Therefore, relief valve testing at low pressure should be minimized to reduce unnecessary damage to the valve seats. Previous testing at 250 psig was performed to ensure that relief valves would function properly at rated pressure if depressurization was required. This testing responded to occurrences where high air supply pressure or incorrectly wired logic circuits resulted in failure of energized solenoids to de-energize and resultant valve opening. For Nine Mile Point - Unit 2, bench tests are performed on each relief valve to provide assurance that each assembly will perform satisfactorily and preoperational tests check out the adequacy of electrical power supply, logic and air supply.

Testing of the ADS valves is required by Plant Technical Specification Surveillance Requirements to ensure that depressurization capability exists. Overpressure protection during transients at power levels below 25% power can also be adequately handled by the safety valves with assistance from the ADS valves. Therefore, single valve testing at Test Condition 1 meets the objective of demonstrating relief valve operability required by Regulatory Guide 1.68, Appendix A, paragraphs 4.p and 5.t.

CONCLUSION:

Testing of the relief valves at Test Condition 1 demonstrates the operability of the relief valves. This proposed testing change does not adversely affect any safety systems or safe operation of the plant and therefore does not involve an unreviewed safety question. Startup Test 26, Relief Valves, can therefore be simplified by replacing the testing at low pressure (250 psig) and between Test Condition 2 and 3 with testing at Test Condition 1 with steam flow greater than the relief valve capacity.

STARTUP TEST 27 - TURBINE TRIP AND GENERATOR LOAD REJECTION
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 5.1.1 and 5.n.n require that a Turbine Trip and Generator Load Rejection be performed at 100% power to demonstrate that the dynamic response of the plant is in accordance with design requirements for turbine trip and full load rejection. These tests may be combined if a turbine trip is initiated directly during the generator load rejection instead of tripping from secondary effects such as a turbine overspeed trip. Startup Test 27, Turbine Trip and Generator Load Rejection, is currently planned to be performed at three conditions during the power ascension test program; (1) a generator load rejection during Test Condition 1 or 2 (within the bypass capacity of the plant); (2) a turbine trip during Test Condition 3 (approximately 75% power); and (3) a generator load rejection at Test Condition 6 (approximately 100% power). It is proposed to delete the Turbine Trip test at Test Condition 3 and change the Generator Load Rejection test at Test Condition 1 or 2 to a Turbine Trip test. This proposed testing will demonstrate that Regulatory Guide 1.68 objectives are met.

DISCUSSION:

Response of the system during a turbine trip and generator load rejection is determined by analyzing test data and comparing to acceptance criteria, Level 1 and Level 2, which define the required system performance. Level 1 criteria require; proper operation of the turbine control and stop valve closure times with respect to the bypass valve opening time, adequate bypass valve response times, proper feedwater control system level response to prevent flooding of the steam lines, that recirculation flow coastdown following protective trips is within design values, and acceptable vessel dome pressure and simulated heat flux response.

Level 2 criteria require; that no MSIV closure occur during the first three minutes of the event, that vessel dome pressure and simulated heat flux changes do not exceed predicted values, for the generator load rejection within bypass capacity (Test Condition 1 or 2) the reactor does not scram, that the bypass capacity calculated is greater than or equal to the assumed value in FSAR analysis, that low water level recirculation pump trip is avoided, that feedwater level control avoids loss of feedwater because of high level trips and require that safety/relief valve discharge temperatures remain within acceptable limits.

The generator load rejection (Test Condition 6) and the turbine trip (Test Condition 1 or 2 within bypass capacity) will provide data to demonstrate that the Level 1 and 2 criteria are met during a turbine trip. The turbine bypass system performance will be verified at a lower power level by changing the proposed generator load rejection at Test Condition 1 or 2 to a turbine trip. Integrated system response to a turbine trip can be obtained from the Generator Load Rejection test at Test Condition 6.

Control systems which regulate the long term operation following the transients are separately tested during the power ascension test program. Feedwater and level control system tuning in Startup Test 23A (Feedwater System Response) will ensure proper water level control. High and low water level trip avoidance will be verified in the Generator Load Rejection test at Test Condition 6.

CONCLUSIONS:

The turbine trip test has been previously demonstrated to be a mild transient event and poses no serious threat to the core and reactor integrity. In addition, the transient results from a generator load rejection at full power are more limiting than the results from a turbine trip at Test Condition 3 and 6. Based on the above discussions, the proposed change will not adversely affect the safety related systems or safe operation of the plant and therefore does not involve an unreviewed safety question.

Current testing of the generator load rejection at 100% power, satisfies the requirements imposed by Regulatory Guide 1.68 (Revision 2), Appendix A, paragraphs 5.1.1 and 5.n.n. In addition, the proposed Turbine Trip test within bypass valve capacity (Test Condition 1 or 2) provides additional verification of the response of the protective systems and also provides demonstration of the bypass system's capability to avoid scram at low power levels. Therefore, the turbine trip at Test Condition 3 can be deleted with the added change that the generator load rejection at Test Condition 1 or 2 (within bypass capacity) will be changed to a turbine trip test.

STARTUP TEST 30E - RECIRCULATION SYSTEM CAVITATION TEST SIMPLIFICATION

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 5.s requires that the recirculation flow control system be calibrated as necessary and its performance verified. At conditions of high flow and low power, both the jet pumps and the recirculation pumps may cavitate. The analytically determined cavitation region of the power-flow map is protected by cavitation interlocks which will run back recirculation flow at low power if the cavitation limits are exceeded. The recirculation system flow control valves (FCV) will cavitate at conditions of low power and low subcooling. Cavitation interlocks are also provided to protect against FCV cavitation and will run back recirculation flow upon sensing a decrease in subcooling (as measured by a low feedwater flow). In both of the above cases, flow runback is caused by a shift in the power supply to the recirculation pump motors from normal power to the low frequency motor generators.

Startup Test 30E, Recirculation System Cavitation, verifies that no recirculation system cavitation will occur in the operable region of the power-flow map. Currently, this test is planned to be performed at Test Condition 2 and 3, where power will be lowered until a recirculation flow runback occurs, and verifying that cavitation has not occurred. It is proposed that the testing be simplified such that operation at the analytically determined cavitation limit is performed to determine if cavitation occurs and if the recirculation runback feature is actuated (however, the interlock will be temporarily bypassed to prevent runback of the recirculation flow).

DISCUSSION:

Response of the system near the cavitation region is determined by analyzing test data and comparing to acceptance criteria which define the required system performance. For the recirculation system cavitation test, the recirculation runback logic is required to have

settings adequate to prevent operation in areas of potential cavitation. This may be accomplished without requiring that recirculation flow runback occur. The recirculation flow runback logic can be bypassed temporarily during the Startup Test 30E. Power can then be reduced by inserting control rods down to the cavitation interlock and verifying that the interlock is actuated (although actual recirculation flow runback will not occur). Signals used to detect pump or FCV cavitation can then be monitored to verify that cavitation has not occurred. With appropriate placement of the jumper on the cavitation interlock, no other recirculation flow runback logic feature will be affected. Should a feedwater transient occur during the performance of this test, the operators can manually runback recirculation flow as necessary to prevent cavitation. The cavitation interlock is not a safety related feature and therefore does not adversely affect the safe operation of the plant.

CONCLUSION:

Cavitation interlocks designed to prevent operation in regions of potential pump or FCV cavitation are analytically determined on a conservative basis. The proposed simplified testing will verify that cavitation does not occur at or above the cavitation interlocks. This proposed testing demonstrates that the acceptance criteria are satisfied and will not adversely affect any safety related systems or safe operation of the plant and does not involve an unreviewed safety question. Therefore, Startup Test 30E, Recirculation System Cavitation, can be simplified as stated above.

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2; August 1978), paragraphs C.1.a and C.1.b require that systems which will be used for shutdown and cooldown of the reactor under normal, transient and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions should be tested to demonstrate that the plant can be operated in accordance with design requirements. In addition, Appendix A, paragraph 5.1 requires that the design capability of the residual heat removal (RHR) system in the steam condensing mode be demonstrated. Startup Test 71, RHR System, demonstrates the ability of the RHR system to remove residual and decay heat from the nuclear system. Currently, testing is planned to be performed in the suppression pool cooling mode, the shutdown cooling capacity mode and in the steam condensing mode. It is proposed to defer obtaining heat exchanger data in the shutdown cooling mode until after commercial operation.

DISCUSSION:

The RHR system is designed to remove residual and decay heat generated by the core under normal and abnormal shutdown conditions. The suppression pool cooling function of the RHR system ensures that the temperature in the suppression pool immediately following blowdown does not exceed a predetermined limit. The shutdown cooling function of the RHR system removes residual and decay heat from the nuclear boiler system after reactor shutdown in preparation for refueling or nuclear system servicing. Testing of these two functions demonstrates compliance to Regulatory Guide 1.68, paragraphs C.1.a and C.1.b. During nuclear boiler system isolation, steam at reduced pressure and temperature can be directed from the main steam lines to the RHR heat removal system heat exchanger when the RHR system is operated in the steam condensing mode. This function of this mode of operation is to increase plant availability

by maintaining hot standby conditions until a plant restart can be performed.

During the startup test program, the decay heat level is insignificant and can lead to impractical testing of the shutdown cooling mode. Testing of the shutdown cooling mode during the startup test program may result in exceeding the 100°F/hour cooldown rate limit for the reactor pressure vessel (RPV) if both RHR heat exchangers are used for the test. The shutdown cooling mode capacity test (heat exchanger data) can be deferred until sufficient decay heat is available (sometime after completion of the Power Ascension Test Program) without significantly affecting plant safety because the mode is not required to provide cooling during accident conditions and the requirements of 10CFR50, Appendix A, General Design Criteria 34 (Residual Heat Removal) are met by the RHR alternate shutdown cooling mode (i.e., suppression pool cooling mode). In addition, shutdown cooling mode heat exchanger capacity may be inferred from data taken in the suppression pool cooling mode.

CONCLUSION:

Testing of the suppression pool cooling and steam condensing mode of operation of the RHR system meets the objectives of Regulatory Guide 1.68. Shutdown cooling mode testing may result in exceeding RPV cooldown limits because of insufficient decay heat. The shutdown cooling mode of operation is not safety related and the suppression pool cooling mode provides an alternate shutdown cooling mode in addition to data which can be used to infer shutdown cooling capacity. Therefore, deferring testing of the shutdown cooling mode heat exchanger capacity until after completion of the power ascension test program does not adversely affect any safety systems or the safe operation of the plant and therefore does not involve an unreviewed safety question.

TEST CONDITION 4 - NATURAL CIRCULATION OPERATION
TEST SIMPLIFICATION - DELETE TEST CONDITION

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 5 requires that "appropriate consideration should be given to testing at the extremes of possible operating modes for facility systems. Testing under simulated conditions of maximum and minimum equipment availability within systems should be accomplished if the facility is intended to be operated in these modes." Test Condition 4 defines the region of the power/flow map on the natural circulation core flow line within ± 5% of the intersection of the 100% rod line. Testing of core performance and control system response is currently planned to be performed in this region. It is proposed to simplify the scheduled testing at Test Condition 4 and include this testing within Test Condition 5. Test Condition 4 could therefore be deleted.

DISCUSSION:

Although operation at natural circulation conditions is not an intended mode of operation for a BWR, the condition can be reached as the result of a moderate frequency event (two recirculation pump trip, 0.25 events/plant-year). Service Information Letter (SIL) Number 380, Revision 1 (Reference 1) recommends that with no reactor coolant system recirculation loops in operation, the operator should immediately initiate actions to reduce thermal power by inserting control rods to or below the 80% rod line using the plants prescribed control rod shutdown insertion sequence. Core monitoring recommendations are given for use by the operator while preparations are made for the restarting of the recirculation pumps. Therefore, testing is required at natural circulation conditions to verify operator procedures and to determine system performance.

Currently, testing at Test Condition 4 includes tests to determine control systems response (pressure regulator and feedwater control

system), core performance, selected process temperatures, recirculation system performance and core stability (control rod and pressure perturbations). The recirculation system performance also includes restarting of the recirculation pumps. It is proposed to delete the control systems tuning/demonstration tests which are scheduled to be performed at Test Condition 4. The control systems are tested at all other test conditions and their performance is adequately demonstrated over the power/flow range expected during normal operation. Therefore, Startup Test 22, Pressure Regulator, and Startup Test 23A, Feedwater System Response, would be deleted from testing at natural circulation on the 100% rod line.

It is also proposed to replace the stability testing scheduled to be performed at Test Condition 4 (Startup Test 21, Core Power-Void Mode and portions of Startup Test 22, Pressure Regulator) with the surveillance recommendations of SIL-380, Revision 1. This Service Information Letter recommends monitoring of the APRM and LPRM detectors when operating at natural circulation conditions. A set of baseline data is required as a measure of the system characteristics, and will be used to determine acceptable plant behavior during later operation. These surveillance recommendations have been provided by General Electric to all utilities (Reference 1) and have been approved for use by the NRC staff (Reference 2). Stability testing of the reactor will be demonstrated in the normal operating range by the testing performed in conjunction with Startup Test 22, Pressure Regulator, at Test Condition 5. In addition to these plant specific tests, significant special testing has been performed at other BWR plants which characterizes the stability performance for BWR's.

The proposed simplified testing at natural circulation would therefore include the following; (1) trip of the recirculation pumps near the rated rod line from Test Condition 5, (2) insertion of control rods to approximately the 80% rod line in accordance with SIL-380, Revision 1, (3) monitoring of steady state recirculation system performance (Startup

Test 30C, Recirculation System), (4) monitoring of APRM and LPRM detectors (SIL-380, Revision 1), (5) monitoring of selected process temperatures (Startup Test 16A, Selected Process Temperatures), (6) monitoring of core performance (Startup Test 19, Core Performance), and (7) restarting of the recirculation pumps. This proposed simplified testing will provide for operating training during natural circulation operation and will adequately characterize the system performance during these conditions.

Finally, it is proposed that the testing at natural circulation core flow be performed as part of Test Condition 5. Since Test Condition 4 is typically performed directly following Test Condition 5, and since it has been proposed to minimize testing at natural circulation, it is logical that the remaining testing at natural circulation be performed as an extension of Test Condition 5. Acknowledgment of a separate Test Condition 4 could therefore be omitted from the Nine Mile Point Unit 2 Startup Test Program.

CONCLUSION:

Testing of the plant systems is performed over a wide range of operating conditions representing the extremes of possible operating modes intended for the plant. In addition, testing is performed at natural circulation, which although it is not an intended mode of operation, can be encountered following a recirculation pump trip event. The proposed simplified testing provides for adequate operator training and demonstration of system characteristics during natural circulation and therefore meets the objectives of Regulatory Guide 1.68, Appendix A, paragraph 5. The proposed test change does not adversely affect any safety system or the safe operation of the plant and therefore does not involve an unreviewed safety question.

TEST CONDITION 4 - NATURAL CIRCULATION OPERATION
TEST SIMPLIFICATION - DELETE TEST CONDITION

REFERENCES:

1. Letter, C.O. Thomas (NRC) to H.C. Pfefferlen (GE), "Acceptance For Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, 'Thermal Hydraulic Stability Amendment to GESTAR II'", April 24, 1985.

2. "BWR Core Thermal Hydraulic Stability," Service Information Letter 380, Revision 1, General Electric Company, February 10, 1984.

ATTACHMENT 3

UPDATED INFORMATION

[illegible]

[illegible]

TABLE 14.2-202

RADIATION MEASUREMENT

Startup Test (SUT-2)

Test Objectives

1. To determine the background radiation levels in the plant environs prior to operation for base data on activity buildup.

2. To monitor radiation at selected power levels to assure the protection of personnel during plant operation.

Prerequisites

The appropriate preoperational tests have been completed; the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

A survey of natural background radiation throughout the
plant site is made prior to fuel loading. Subsequent to
fuel loading, during reactor heatup and at power levels of
25, 60, and 100 percent of rated power, gamma radiation 22
level measurements, and where appropriate, neutron dose rate
measurements are made at specific locations throughout the 22
plant. All potentially high radiation areas are surveyed.

The following tests are performed:

Action	Test conditions
1. Background radiation level survey.	a. Prior to fuel loading.
2. Monitor radiation level periodically during the startup.	b. Fuel loading.
	c. Reactor heatup.
	d. Steady-state operation at 25, 60, and 100 percent of rated power. 22

TABLE 14.2-202 (Cont)

Acceptance Criteria

Level 1:

The radiation doses of plant origin and the occupancy times of personnel in radiation zones are controlled consistent requirements with the ~~guidelines~~ of the standards for protection against radiation outlined in 10CFR20.

Level 2:

22 Not applicable.

3. Pump head, capacity.

4. System flows.

5. Heat-up characteristics, when attainable.

6. Tuning of system controls.

7. Response to simulated safety signals and/or loss of power.

8. Operating times.

This phase of testing verifies the ability of the plant to support fuel load and power operations.

14.2.1.4 Initial Startup Test Phase

The initial startup test phase commences with the receipt of the operating license and the preparation for fuel load and extends through the 100-percent rated power/100-hr warranty demonstration. The initial startup test phase is divided into six testing plateaus: open vessel (including fuel loading), heatup, low power, medium power, full power, and 22
rated power warranty run. Testing performed during this phase of the program ensures that fuel loading is accomplished in a safe manner, confirms the plant design basis, demonstrates, to the extent possible, the plant's ability to withstand anticipated transients and postulated accidents and verifies that the plant can be safely brought to rated power and sustained power operations.

14.2.2 Organization and Staffing

The Unit 2 startup and test organization and interfaces to plant operations, SWEC, General Electric, and other selected NMPC organizations are shown in Figure 14.2-7 and are discussed in the following sections.

The Unit 2 operational organization is discussed in Chapter 13.0. The initial startup test phase is performed under the control of the station superintendent and coordinated by the ~~Reactor Analyst~~ Power Ascension Manager. The responsibilities of the operations organization during the startup and test program are discussed in the following sections.

Staffing levels during the startup and test program will be commensurate with schedule and project needs and requirements.

14.2.2.3 Station Staff Responsibilities During the Startup and Test Program

14.2.2.3.1 Plant Operations

Plant Operations consists of those personnel who operate Unit 2 under the direction of the ~~Supervisor~~ Superintendent Operations, as described in Chapter 13. This group is responsible for the operation of plant equipment and systems during the startup and test program.

The startup test program is implemented by the plant operations department using procedures developed and approved in accordance with the APs. These procedures are prepared by members of the station staff and others as required under the direction of the Station Superintendent. Technical expertise from other organizations and GE is used whenever necessary.

14.2.2.3.2 Station Support Staff

The station support staff consists of those station personnel who maintain Unit 2. Duties and general responsibilities are provided in Chapter 13. The station 22
support staff, under the technical direction of the Station Superintendent, supports the startup and test program by maintaining all plant equipment, systems, and structures after release to NMPC and by providing technical assistance and manpower support to the extent practical.

The station staff assumes complete control and responsibility for the operation, testing and maintenance of Unit 2 at fuel load.

The ~~Reactor Analyst~~ Power Ascension Manager is responsible to the Station Superintendent for ensuring that all startup test phase procedures are written, reviewed, and approved; coordinating startup test phase testing; ensuring proper documentation of the startup test phase testing; and maintaining test results.

14.2.2.4 General Electric Company

14.2.2.4.1 Site Operations Manager

The GE Site Operations Manager (SOM) is the senior NSSS vendor representative onsite at or near fuel loading and is the official GE spokesman for preoperational and startup testing concerns and requirements. He coordinates with the

Station Superintendent and the Startup Manager for the performance of his duties, which include the following:

1. Review of all NSSS test procedures, including changes and results.

2. Acts as liaison with GE on testing matters involving GE NSSS-supplied equipment.

3. Provide administrative support and supervision to GE onsite personnel involved in the test program.

4. Represent GE on the JTG.

14.2.2.4.2 ok ~~Operations~~ Superintendent

The GE Operations Superintendent is responsible to the GE SOM for the administrative and technical supervision of GE shift superintendents. The Operations Superintendent works directly with the NMPC ~~Supervisor Operations~~ Superintendent Operations and provides GE technical support to the operating organization.

22 14.2.2.4.3 ~~Shift Superintendents~~ Startup Test Operations (STO) Engineer

The GE ~~Shift Superintendents~~ STO Engineers provide technical support to the Unit 2 shift operations personnel in the testing and operation of GE supplied systems. They provide 24-hr day shift coverage as required, beginning with fuel loading, and report to the GE Operations Superintendent.

14.2.2.4.4 Lead Engineer Startup Test, Design and Analysis (STD&A)

The GE Lead Engineer STD&A is responsible to the GE SOM for supervising the GE STD&A engineers ~~verifying core physics parameters and characteristics~~ and documenting that the performance of the NSSS and its components conform to acceptance criteria. He works with the ~~Reactor Analyst~~ Power Ascension Manager or his representative to coordinate and effect implementation of the startup test program, including any special testing required to confirm these acceptance criteria.

14.2.2.4.5 GE Startup, Test, Design and Analysis Engineers

The GE STD&A engineers assist in the execution of the initial startup test phase.

Reverification of Prerequisites

Following a test interruption which results in a halt in testing during a preoperational test, the Test Director shall review the test prerequisites for possible reverification. The results of the review entered in the test summary and any reverified prerequisites listed.

Preoperational Test Summary

A preoperational test summary shall be prepared by the Test Director for all preoperational test procedures. The test summary includes significant events during the test, a description of any problems found during the test and reference to their resolution, any reverification of prerequisites required and an evaluation of the test results with reference to the acceptance criteria. This shall in particular note if any acceptance criteria have not been met. The test summary is attached to the record copy of the test procedure.

14.2.4.3 Initial Startup Testing

Startup testing is conducted by personnel from plant operations, startup and test, GE, and groups as required under the direction of the on-duty SSS in accordance with the APs. 22

During this phase, plant operating procedures are utilized in conjunction with the approved test procedures.

The final authority to start, continue, or end a test is the responsibility of the SSS after all required approvals have been obtained.

The master tracking system is used to ensure that prerequisites for initial fuel loading and the beginning of initial startup testing are fulfilled. In addition, each individual startup test procedure specifies prerequisites that must be validated prior to test performance. The on-duty SSS and respective test personnel ensure that all prerequisites are satisfied prior to performance of any initial startup test.

14.2.4.4 Modifications to Test Procedures During Testing

14.2.4.4.1 Preliminary Tests

Due to their nondetailed nature, field revision of generic preliminary tests is not applicable.

The master tracking system may be replaced by a similiar tracking system as the preoperational test phase nears completion.

During performance, any changes to a specific preliminary test procedure that changes the intent, scope, or acceptance criteria of the test are made on a field revision form (FRF) prior to implementation of the change in accordance with the SAPs.

14.2.4.4.2. Preoperational Tests

During performance, no changes may be made to the procedure that change the intent, scope or acceptance criteria of the test without the prior approval of the JTG. These changes are made on an FRF in accordance with the SAPs. The Test Director may elect to perform unaffected sections of the test while awaiting resolution from JTG, provided test sequence is not mandatory.

Other exceptions and minor corrections to the test procedure are authorized in accordance the SAPs.

14.2.4.4.3 Startup Tests

Modification to initial startup test procedures are
22 classified as major or minor changes. A major change changes the intent of the procedure and requires development of a revision to the procedure. Such a revision requires approval of the organizations that originally approved the test procedure. When a procedure in progress cannot be followed or completed and a major change to the procedure is required, the test is held at that point, the system placed in a stable condition, and the necessary approvals obtained in accordance with the APs prior to continuing the test.

Minor changes do not change the intent of the test procedure and may be made with the concurrence of the ~~duty SSS~~ an SRO Licensed member of the plant staff at the time the test is run. Minor changes to procedures are made in accordance with the APs which detail the method of entry of the change and the required approvals.

14.2.4.5 Modifications and Deficiencies

14.2.4.5.1 Preliminary and Preoperational Phases

The SAPs contain administrative controls for identifying, reporting, and tracking of deficiencies and modifications during these phases.

Changes to plant system and equipment design are reviewed and approved in the same manner as the original design by the approved design organization.

auxiliary power and the capability of the offsite power system to supply power to start and run emergency core cooling and selected normal loads during a simulated LOCA condition.

14.2.10.1.2 Cold Functional Testing

Cold functional testing is defined as an integrated system operation of various plant systems that can be operated prior to fuel loading. The intent is to observe any unexpected operational problems from either an equipment or a procedural standpoint and to provide an opportunity for further operator familiarization with the system operating procedures under operating conditions.

Some cold functional testing is accomplished during the preoperational test program. For example, integrated and simultaneous operation of the following systems may take place duirng the flush of the total system: condensate system, condensate demineralizer system, low pressure coolant injection (LPCI) system, core spray systems, reactor water cleanup (RWCU) system, service water systems, turbine
22 building closed cooling water (TBLCW) system, reactor building closed cooling water (RBCLCW) system, and others. As required, additional integrated system performance will be demonstrated prior to fuel loading.

14.2.10.1.3 Routine Surveillance Testing

Because the interval between completion of a preoperational test on a system and system operation may be of considerable length, a number of routine surveillance tests must be performed prior to fuel loading and must be repeated on a routine basis. The technical specifications (Chapter 16) detail the test frequency. In general, this surveillance test program is instituted prior to fuel loading by the plant operating staff.

14.2.10.1.4 Master Tracking System (MTS)

A detailed list of items that must be completed, including work requests, design changes, and proper disposition of all exceptions noted during preoperational testing listed in Table 14.2-1, is rechecked to verify completion prior to the final approvals for fuel loading and for those items required, at each significant new step such as heatup, opening main steam isolation valves (MSIVs), and turbine generator operation. The MTS may be replaced by a similar tracking system as the preoperational test phase nears completion.

14.2.10.1.5 Initial Fuel Loading (Open Vessel Plateau)

Fuel loading requires the movement of the full core complement of assemblies from the fuel pool to the core, with each assembly identified by number before being placed in the correct coordinate position. The procedure controlling this movement is arranged so that operability checks of installed neutron instrumentation are made at predetermined intervals throughout the loading, thus demonstrating reliable monitoring capability to ensure subcriticality is maintained throughout fuel loading. A complete check is made of the fully loaded core to ascertain that all assemblies are properly installed, correctly oriented, and occupying their designated positions.

14.2.10.1.6 Zero Power Level Tests (Open Vessel Plateau)

At this point, a number of tests are conducted that are best described as initial zero power level tests. Chemical and radiochemical tests are made in order to check the quality of the reactor water before and after fuel loading and to establish base and background levels that are required to facilitate later analysis and instrument calibrations. 22
Plant and site radiation surveys are made at specific locations for comparison with the values obtained at the subsequent operating power levels. ~~Shutdown Margin~~ Control rod drive system testing takes place while the reactor vessel is assembled in preparation for initial criticality and initial heatup.
~~at maximum gain.~~

See next page

14.2.10.2 Initial Criticality and Heatup to Rated Temperature and Pressure

Initial criticality and Heatup follows the satisfactory completion of the fuel loading and zero power level tests (Sections 14.2.10.1.5 and 14.2.10.1.6). ~~and~~ Further checks are made of coolant chemistry together with radiation surveys at the selected plant locations. All CRDs are scram-timed at rated pressure prior to initial heatup. The process computer checkout continues as more process variables become available for input. The reactor core isolation cooling (RCIC) system will undergo controlled starts at low reactor pressure and at rated conditions, with testing in the quick-start mode at rated pressure ~~1,000 psig~~. Correlations are obtained between reactor vessel temperatures at several locations and the values of other process variables as heatup continues. The movements of NSSS piping in the drywell, mainly as a function of expansion are recorded for comparison with design data.

TECHNICAL REVIEW (LEAD STD&A AND ~~(B&W ANALYST)~~ Power Ascension Manager)

↓

STAFF REVIEW (IN ACCORDANCE WITH AP'S)

↓

SORC REVIEW AND ACCEPTANCE

↓ - - - → SRAB REVIEW*

APPROVAL BY GENERAL SUPERINTENDENT

*ONLY AS REQUIRED BY THE GENERAL SUPERINTENDENT

FIGURE 14.2-2

REVIEW CYCLE FOR INITIAL STARTUP TEST PROCEDURES

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
FINAL SAFETY ANALYSIS REPORT

TECHNICAL REVIEW
AND ACCEPTANCE (GE LEAD STD&A, ~~AND~~ S.O.M., Technical Supt., and Power Ascension Manager)

↓

STATION SUPERINTENDENT
REVIEW AND ACCEPTANCE

↓

QUALITY ASSURANCE
REVIEW AND ACCEPTANCE

↓

SORC
REVIEW AND ACCEPTANCE

↓ - - - → SRAB*

GENERAL SUPERINTENDENT
REVIEW AND APPROVAL

*ONLY AS REQUIRED BY THE GENERAL SUPERINTENDENT

FIGURE 14.2-4

INITIAL STARTUP TEST
RESULTS REVIEW

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT UNIT 2
FINAL SAFETY ANALYSIS REPORT

TABLE 14.2-203 8

FUEL LOADING 8

Startup Test (SUT-3)

Test Objective

To load fuel safely and efficiently to the full core size.

Prerequisites

Prerequisites to fuel loading are established in Section 14.2.10 and the tests required thereby are implied in those prerequisites. Also, the SORC has approved fuel loading and the following additional prerequisites have been met to assure that fuel loading is performed in a safe manner:

1. All systems required for fuel loading have undergone preoperational testing.

2. Fuel and control rod inspections are complete. Control rods are installed and tested.

3. SRMs are calibrated and operable. IRMs and APRMs have been preoperationally tested and are operable.

4. SRMs are source checked with a neutron source prior to loading and periodically will be functionally checked and source checked.

5. The status of the reactor building is specified and established.

6. Reactor vessel status is specified relative to internal component placement and this placement established to make the vessel ready to receive fuel.

7. Reactor vessel water level is established and minimum level prescribed.

8. The standby liquid control system is operable.

TABLE 14.2-203 (Cont)

Test Procedure

1. Prior to fuel loading, control rods and neutron sources and detectors are installed and tested. Fuel loading will begin ~~around~~ near an SRM location. Loading will spiral outward from this location to the fully loaded
22 configuration. A shutdown margin demonstration on the partially loaded core is performed during the loading.

Acceptance Criteria

Level 1:

The partially loaded core shutdown margin demonstration
22 verifies that the configuration is subcritical by at least 0.38-percent $\Delta K/K$ with the analytically determined strongest rod fully withdrawn.

Level 2:

22 Not applicable

TABLE 14.2-207

SOURCE RANGE MONITOR PERFORMANCE

Startup Test (SUT-6)

Test Objective

To demonstrate that the operational sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner.

Prerequisites

The preoperational tests have been completed, and the SORC has reviewed and approved the test procedure and the initiation of testing. The CRD system must be operational.

Test Procedure

Prior to fuel load commencement, SRM operability will be demonstrated as part of Startup Test 3 and the Technical Specifications. Prior to rod withdrawal to initial criticality, this test will also demonstrate that SRM signal-to-noise and minimum count rate are in accordance with Technical Specification requirements. During rod withdrawal to initial criticality, SRM response will be monitored and used to determine the proximity to criticality. This rod withdrawal sequence shall specify all control rod withdrawals from all-rods-in to rated power. Once the reactor is critical, proper overlap of the SRMs with the IRMs will be demonstrated.

The following tests are performed:

Action	Test Conditions*
1. Rod withdrawal in prescribed sequence	a. After fuel loading. b. Operational neutron sources installed. c. SRM minimum signal-to-noise count ratio and minimum count rate criteria satisfied.
2. Verify SRM-IRM overlap	a. Flux level sufficient for IRM response.

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

TABLE 14.2-207 (Cont)

Acceptance Criteria

Level 1:

1. There is a neutron signal-to-noise count ratio of at least 2 to 1 on the required operable SRMs or fuel loading chambers.

2. Minimum count rate is in accordance with the technical specifications.

3. Each IRM channel must be on scale before the SRMs exceed their rod block setpoint.

Level 2:

Not applicable.

TABLE 14.2-208 (Cont)

The following test is performed:

Action		Test Conditions*
Demonstrate the rod sequence adjustment procedure	a.	Reduce recirculation flow.
	b.	Sufficient margin available to PCIOMR envelope and core operating limits.

Acceptance Criteria

Level 1:

Completion of the adjustment of one rod pattern for the complementary pattern with continual satisfaction of all licensed core limits constitutes satisfaction of the requirements of this procedure.

Level 2:

All nodal powers will remain below their PCIOMR threshold limit during this test.

* This test will be performed during the first regularly scheduled control rod sequence exchange during the first fuel cycle.

TABLE 14.2-209

INTERMEDIATE RANGE MONITOR PERFORMANCE

Startup Test (SUT-10)

Test Objective

To adjust the IRM system to obtain an optimum overlap with the SRM and APRM systems.

Prerequisites

The preoperational tests have been completed. The SORC has reviewed and approved the test procedures and the initiation of testing. All SRMs and pulse preamplifiers, IRMs and voltage preamplifiers, and APRMs have been calibrated in accordance with vendor's instructions.

Test Procedure

Initially the IRM system is set to maximum gain. After the APRM calibration, the IRM gains are adjusted to optimize the IRM overlap with the SRMs and APRMs.

The following tests are performed:

Action	Test Conditions
1. Verify IRM-SRM overlap.	Flux level sufficient for IRM response.
1 ~~2~~. Verify IRM response to neutron flux.	Flux level sufficient for IRM response.
2 ~~3~~. Adjust IRM gain, if necessary, for proper IRM-APRM overlap.	During first APRM calibration based on a heat balance.

Acceptance Criteria

Level 1:

a) Each IRM channel must be on scale before the SRMs ~~exceed~~ their rod block setpoint. 12

a ~~b)~~ Each APRM must be on scale before the IRMs exceed their upscale rod block setpoint.

TABLE 14.2-209 (Cont)

22 b ~~c)~~ The IRMs produce a scram as specified in the Technical Specifications with the reactor mode switch in startup.

Level 2:

Each IRM channel must be adjusted so a half-decade overlap with the SRMs and one-decade overlap with the APRMs ~~are~~ is assured.

TABLE 14.2-210

LPRM CALIBRATION

Startup Test (SUT-11)

Test Objective

To calibrate the LPRM system.

Prerequisites

The appropriate preoperational tests have been completed, the SORC has reviewed and approved the test procedures and the initiation of testing. Instrumentation for calibration has been checked and installed.

Test Procedure

The LPRM channels are calibrated to make the LPRM readings proportional to the neutron flux in the LPRM water gap at 22 the chamber elevation. Calibration factors are obtained through the use of either an offline or a process computer calculation that relates the LPRM reading to average fuel assembly power at the chamber height.

The following tests are performed:

Action	Test Conditions	
1. Verify LPRM flux response. This test may be done in conjunction with rated pressure scram testing (SUT-5).	a. Hot standby or TCI.	22
2. Take data and calibrate LPRM system.	a. TC-2, 3 and TC-6. b. All systems in NORM mode.	22

8 TABLE 14.2-210 (Cont)

Acceptance Criteria

Level 1:

Not applicable

Level 2:

Each LPRM reading is within 10 percent of its calculated value.

TABLE 14.2-211

APRM CALIBRATION

Startup Test (SUT-12)

Test Objective

To calibrate the APRM system.

Prerequisites

The preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and the initiation of testing. Instrumentation for calibration has been checked and installed.

Test Procedure

A heat balance is ~~generally made each shift~~ performed daily and after each major power level change. Each APRM channel reading is adjusted to be consistent with the core thermal power as determined from the heat balance in accordance with the Technical Specifications. During heatup a preliminary calibration is made by adjusting the APRM amplifier gains so that the APRM readings agree with the results of a constant heatup rate heat balance. The APRMs are recalibrated in the power range by a heat balance as soon as adequate feedwater indication is available.

Recalibration of the APRM system is in accordance with the technical specifications. 22

The following tests are performed:

Action	Test Conditions
1. Calibrate APRM system based on heat balance data.	Constant rate of heatup below rated pressure.
2. Calibrate APRM system based on steady-state heat balance data.	Above Approximately 25 percent power at TC-2, 3, 5, and 6 and repeated as necessary.

TABLE 14.2-211 (Cont)

Acceptance Criteria

Level 1:

1. The APRM channels are calibrated to read equal to or greater than the actual core thermal power.

2. Technical specification limits on APRM scram and rod block are not exceeded.

3. In the startup mode, all APRM channels produce a scram at less than or equal to 15 percent of rated thermal
12 power.

Level 2:

If the above criteria are satisfied, then the APRM channels will be considered to be reading accurately if they agree with the heat balance or the minimum value required based on
12 peaking factor, MLHGR, and fraction of rated power to within (+~~1~~, -0) percent of rated power.

+2

APRM upscale scram and rod block trip Set points do not exceed the allowable Values specified in the Technical Specifications.

TABLE 14.2-215

WATER LEVEL REFERENCE LEG TEMPERATURE

Startup Test (SUT-16B)

Test Objective

To measure the reference leg temperature and recalibrate the instruments if the measured temperature is different from the value assumed during the initial calibration.

Prerequisites

The preoperational tests have been completed, the SORC has reviewed and approved the test procedures and initiation of testing. System and test instrumentation have been calibrated.

Test Procedures

To monitor the reactor vessel water level, five level instrument systems are provided. These systems and their functions are: 22

1. Shutdown range - water level measurement in cold, shutdown condition.

2. Narrow range - feedwater flow and water level control functions.

3. Wide range - safety functions.

4. Fuel range - post accident indication. 22

5. Upset range - water level measurement during transient conditions.

The test ~~is done at rated~~ temperature and pressure and under steady-state conditions and ~~verifies that~~ the ~~reference~~ leg temperature of the instrument ~~is the~~ value assumed during initial calibration~~.~~ ~~If~~ not, the instruments ~~are~~ ~~recalibrated~~ using the measured value.

Action	Test Conditions
Monitor drywell temperature.	Hot standby with steady drywell temperatures.

The test for the narrow range, wide range and upset range level instruments will be done during steady state conditions at rated temperature and pressure. The test for the shutdown range level instrument will be done during cold ambient conditions with the reactor shutdown. No test is possible for the fuel zone water level instrument by virtue of its calibration conditions(ie LOCA conditions). The testing will verify that the reference leg temperature of the instrument is the value assumed during calibration. If not, the instruments will be recalibrated using the measured reference leg temperature.

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TABLE 14.2-215 (Cont)

<u>Acceptance Criteria</u>

Level 1:

Not applicable.

Level 2:

The difference between the actual reference leg temperature(s) and the value(s) assumed during initial
12 calibration shall be less than that amount which will result in a scale end point error of 1 percent of the instrument span for each range.

TABLE 14.2-216

SYSTEM EXPANSION

Startup Test (SUT-17)

Test Objectives

To demonstrate that:

1. The piping system during system heatup and cooldown is free to expand and move without unplanned obstruction or restraint.

2. The piping does shake down after a few thermal expansion cycles.

3. The measured values of displacement are within the limits specified by the responsible piping design engineer.

Prerequisites

The appropriate preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and initiation of testing.

Test Procedure

~~Hanger positions of reactor recirculation piping are recorded after each major thermal cycle until a shakedown has taken place (normally about three cycles).~~ During initial heatup, a visual inspection is made at an intermediate reactor water temperature to ensure that components are free to move as designed. Corrections are made as necessary. Devices for continuously measuring pipe displacement are mounted on the recirculation lines, and motion during heatup is compared with calculated values. 22
Final sensor locations are determined at the site and based on generic recommendations. After receipt of the installed transducer locations, the plant piping design subsection will supply to the startup engineer the expected thermal displacements (Level 2 limits) and the allowable thermal displacements (Level 1 limits) for the above piping and related conditions. These displacements will be specific to each transducer for each coordinate direction.

TABLE 14.2-216 (Cont)

Action	Test Conditions
1. Visual inspection ~~and hanger readings.~~	a. All control systems in NORM mode. b. Approximately ~~250°F~~ 275°F at accessible locations. c. At ambient and rated temperature. ~~d. After 3 to 5 complete heatup and cooldown cycles.~~
2. Record displacement sensor readings.	a. At approximately ~~250°F~~ 275°F. ~~b. At rated recirculation temperature.~~ 22 ~~c. At rated feedwater temperature.~~

Acceptance Criteria (as described in response to Question F210.37)

Level 1:

1. There shall be no obstructions which will interfere with the thermal expansion of the ~~main steam and~~ recirculation piping systems.

22 2. The displacements at the established transducer locations shall not exceed the allowable values provided ~~later~~ by the plant piping design subsection. The allowable values of displacement shall be based on not exceeding ASME Section III Code stress allowables.

Level 2:

The displacements at the established transducer locations shall not exceed the expected values provided ~~later~~ by the plant piping design subsection.

b. At approximately 400 to 450 °F
c. At approximately rated recirculation temperature.
d. Repeat a, b and c for approximately two to four heatup and cooldown cycles

TABLE 14.2-217

~~TRAVERSING INCORE PROBE~~
TIP UNCERTAINTY ~~CALCULATIONS~~

Startup Test (SUT-18)

Test Objective

To determine the reproducibility of the TIP system readings.

Prerequisites

System installation has been completed and preoperational tests completed and verified. The SORC has reviewed and approved the test procedure and initiation of testing. The TIP detector, ball valve time delay, core top and bottom 22
limits, clutch, x-y recorder, and purge system are operational. Instrumentation has been calibrated and installed.

Test Procedure

TIP reproducibility consists of a random noise component and a geometric component. The geometric component is due to a variation in the water gap geometry and TIP tube orientation from TIP location to location. Measurement of these components is obtained by taking repetitive TIP readings at a single TIP location, and by analyzing pairs of TIP readings taken at TIP locations that are symmetrical about the core diagonal of fuel loading symmetry.

TIP data is taken at ~~TC-3 and again at near rated power.~~ greater than 70% rated thermal power. 22
The TIP data are taken with the reactor operating with an octant symmetric rod pattern and at steady-state conditions.

The total TIP reproducibility is obtained by dividing the standard deviation of the symmetric TIP pair nodal ratios by $\sqrt{2}$. The nodal TIP ratio is defined as the nodal BASE value of the TIP in the lower right half of the core divided by its symmetric counterpart in the upper left half. The total TIP reproducibility value that is compared with the test criterion is the average value of the data sets taken.

The random noise uncertainty is obtained from successive TIP runs made at the common hole, with each of the TIP machines making six runs. The standard deviation of the random noise is derived by taking the square root of the average of the

TABLE 14.2-217 (Cont)

variances at nodal levels 5 through 22, where the nodal variance is obtained from the fractional deviations of the successive TIP values about their nodal mean value.

The geometric component of TIP reproducibility is obtained by statistically subtracting the random noise component from the total TIP reproducibility.

The following test is performed:

Action	Test Conditions
TIP overall uncertainty.	a. Octant symmetric control rod pattern. b. At steady state. c. ~~TC-3 and 6.~~ Greater than 70% rated thermal power

Acceptance Criteria

Level 1:

Not applicable.

Level 2:

The total TIP uncertainty (including random noise and geometrical uncertainties) obtained by averaging the
12 uncertainties for all data sets shall be less than 6.0
percent.

TABLE 14.2-219 8

STEAM PRODUCTION

Startup Test (SUT-20) 8

Test Objective

To demonstrate that the NSSS provides steam sufficient to satisfy all appropriate warranties as defined in the contract.

Prerequisites

Test Procedure

Warranty demonstration consists of recording sufficient data under steady-state conditions to determine the reactor power level, the pressure and quality of the steam, and the steam flow rate from the reactor.

These measurements include the temperature, pressure, and flow rate of feedwater entering the reactor, the energy added to the reactor water by the recirculation drive pumps, the flow rate through and temperature entering and leaving the reactor cleanup system, the flow rate and temperature of the CRD cooling water, the carryover of reactor water into the steam lines, and the steam pressure outside the drywell near the MSIVs.

Each set of measurements is taken at frequent intervals, every 5 or 10 min as appropriate, for a total test run duration of 4 hr. The average measured quantity, suitably corrected for all calibration factors, is used to determine the NSSS output during the test run. Where the contract requires a 100-hr demonstration, two test runs are made, one in the first 50 hr and one in the second 50 hr. The demonstrated output is the average of the values from the two test runs. During the balance of the 100-hr demonstration, the NSSS output is held constant within ±5 percent of the nominal steam flow rate as indicated by the installed plant feedwater instrumentation.

Should the 100-hr warranty run be interrupted once for any reason and a subsequent time for any reason not due to the fault of the customer, subject to the provisions of the

Table 14.2-219 (Cont)

contract, it will be repeated. If the test is interrupted a second or subsequent time for any reason due to the fault of the customer or the power grid to which the station is connected, it will be resumed upon coming to full power and continued until the desired test period is accumulated, provided that the minimum continuous period full-power operation has been 24 hr.

The following test is performed:

<u>Action</u>	<u>Test Conditions</u>
Demonstrate steam quality and flow under steady conditions.	a. At conditions prescribed in the nuclear steam system warranty (TC-6). b. Operate continuously for 100 hr.

<u>Acceptance Criteria</u>

Level 1:

1. The NSSS parameters as determined by using normal operating procedures are within the appropriate license restrictions.

~~2.~~ 1. The NSSS is capable of supplying steam in an amount and quality corresponding to the final feedwater temperature and other conditions shown on the rated steam output curve in the NSSS technical description. The rated steam output curve provides the warrantable reactor vessel steam output as a function of feedwater temperature, as well as warrantable steam conditions at
22 the outboard MSIVs. Thermodynamic parameters are consistent with the 1967 ASME steam tables. Correction techniques for conditions that differ from the contracted conditions will be mutually agreed to prior to the performance of the test.

22 Level 2:

~~Not applicable.~~

TABLE 14.2-227

MAIN STEAM ISOLATION VALVES
FUNCTIONAL TESTS

Startup Test (SUT-25A)

Test Objectives

1. To functionally check the MSIVs for proper operation at selected power levels.

2. To determine isolation valve closure time at rated conditions.

3. To determine maximum power at which a single valve closure can be made without scram.

Prerequisites

The preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

(Valve stroke complete)

At 5 percent and greater power levels, individual fast closure of each MSIV will be performed to verify their functional performance and to determine closure times. The times to be determined are: a) the time from the initiation signal to deenergize the solenoids until the valve is stroked from the open position to completely closed (t_{sol}), and 2) the valve stroke time (t_s). Time t_s equals the interval from when the valve starts to move from full open until it is 100 percent closed (valve stroke complete).

To determine the maximum power level at which full individual closures can be performed without a scram, the first MSIV actuations will be performed between 40- and 55-percent power. The results of the tests at 40 to 55-percent power will be used to extrapolate to the next power test point, which will be between 60- and 85-percent power. The test results will ultimately be used to determine the maximum power test condition that has ample margin to scram.

TABLE 14.2-227 (Cont)

The following tests are performed:

Action	Test Conditions
1. Individually close each MSIV, fast mode.	a. Heatup and between TC-1 and -3, close each MSIV to measure valve timing only. b. Recirculation system in POS mode; other systems in NORM mode.
2. Close fastest MSIV, fast mode.	a. Close one valve between 40 and 55 percent power (TC-2 or 3) and again between 60 and 85 percent power (TC-3 or 5). Perform third test at chosen maximum power condition for all subsequent surveillance tests. b. Recirculation system in POS mode at TC-2 and 3 and FLX mode at TC-5. Other systems in NORM mode.

Acceptance Criteria

Level 1:

The MSIV stroke time (t_s) shall be no faster than 3.0 seconds (average of the fastest valve in each steam line), and for any individual valve 2.5 seconds $\leq t_s \leq 5$ seconds. Total effective closure time for any individual MSIV shall be t_{sol} plus the maximum instrumentation delay
25 time ~~, as determined during the Preoperational Test Program~~ and shall be ≤5.5 seconds.

TABLE 14.2-227 (Cont)

Level 2:

1. The reactor shall not scram. The peak neutron flux must be at least 7.5 percent below the trip setting. The peak vessel pressure must remain at least 10 percent psi below the high-pressure scam setting. The peak simulated heat flux must be 5 percent less than its trip point.

2. The reactor shall not isolate. The peak-steam flow on each line must remain 10 percent below the high-steam flow isolation trip setting. 22

TABLE 14.2-228

FULL REACTOR ISOLATION

Startup Test (SUT-25B)

Test Objective

To determine the reactor transient behavior that results from the simultaneous full closure of all MSIVs.

Prerequisites

The preoperational tests have been completed; the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

A test of the simultaneous full closure of all MSIVs is performed at ≥95 percent of rated thermal power. Correct performance of the RCIC, HPCS, and relief valves is shown. 12
Reactor process variables are monitored to determine the transient behavior of the system during and following main steam line isolation.

The following test is performed:

Action	Test Conditions	
Close all MSIVs (SUT-~~33~~ 77 and SUT-5 are to be done in conjunction with this test).	a. Perform at TC-6. b. All systems in NORM mode.	12

Acceptance Criteria

Level 1:

1. Reactor must scram to limit the severity of the neutron flux and simulated fuel surface heat flux transient.

2. Feedwater system settings must prevent flooding of the steam lines.

3. The recorded MSIV full closure times must meet the previously stated timing specifications (SUT-25A).

TABLE 14.2-228 (Cont)

4. The positive change in vessel dome pressure occurring within the first 30 sec after a closure of all MSIV valves must not exceed the Level 2 criteria by more than 25 psi. The positive change in simulated heat flux must not exceed the Level 2 criteria by more than 2 percent of the rated value.

22

Level 2:

1. The positive change in vessel dome pressure and simulated flux occurring within the first 30 sec after the closure of all MSIV valves must not exceed the BOL predicted values. Predicted values will be referenced to actual test conditions of initial power level and dome pressure and will use BOL nuclear data.

2. Initial action of the RCIC and HPCS are automatic when Level 2 is reached, and system performance is within specifications.

3. Recirculation pump trip shall be initiated if low water level (L2) is reached. Recirculation pump power will shift to the low frequency motor generators if low water level (L3) is reached.

4. The temperature measured by thermocouples on the discharge side of the safety/relief valves must return to within 10°F of the temperature recorded before the valve was opened.

TABLE 14.2-231 (Cont)

within 0.3 sec from the beginning of control or stop valve closure motion.

2. Feedwater system settings must prevent flooding of the steam line following these transients.

3 excluding the first 0.25 seconds

3. The two pump drive flow coastdown transients during the first 5 sec must be bounded by the criteria that are specified in SUT-30B. 22

4. The positive change in vessel dome pressure occurring within 30 sec after either generator or turbine trip must not exceed the Level 2 criteria by more than 25 psi.

5. The positive change in simulated heat flux must not exceed the Level 2 criteria by more than 2 percent of rated value.

6. The total time delay from start of turbine stop valve motion or from start of turbine control valve motion to the complete suppression of electrical arc between the fully open contacts of the RPT circuit breakers shall be less than 190 milliseconds. 22

Level 2:

1. There shall be no MSIV closure during the first 3 min of the transient and operator action shall not be required during that period to avoid the MSIV trip. (The operator may take action as he desires after the first 3 min, including switching out of run mode. The operator may also switch out of run mode in the first 3 min if he confirms from measured data that this action did not prevent MSIV closure.) 22

2. The positive change in vessel dome pressure and in simulated heat flux which occurs within the first 30 sec after the initiation of either generator or turbine trip must not exceed the predicted values.

See Next page

(Predicted values are referenced to actual test conditions of initial power level and dome pressure and use beginning-of-life nuclear data. Worst-case design or technical specification values of all hardware performance are used in the prediction, with the exception of control rod insertion time and the delay

TABLE 14.2-231 (Cont)

start note on page 3 of 4

~~from beginning of turbine control valve or stop valve motion to the generation of the scram signal. The predicted pressure and heat flux are corrected for the actual measured values of these two parameters.)~~

3. For the ~~generator~~ turbine trip within the bypass valve capacity, the reactor must not scram for initial thermal power values within that bypass valve capacity and below the
22 power level of which trip scram is inhibited. The measured bypass capacity (in percent of rated power) is equal to or greater than that used for FSAR analysis.

4. Low level initiation of total recirculation trip, HPCS, and RCIC must not occur.

5. Recirculation LFMG sets must take over after the initiation of RPT and adequate vessel temperature difference must be maintained.

6. Feedwater level control must avoid loss of feedwater due
22 to possible high level (L8) trip during the event.

7. The temperature measured by thermocouples on the discharge side of the safety/relief valves must return to within 10°F of the temperature recorded before the valve was opened.

The predicted values are those specified in the Transient Safety Analysis and Design Report. The predicted values should be corrected for actual plant parameters measured during the Startup Test Program.

TABLE 14.2-232

SHUTDOWN FROM OUTSIDE THE MAIN CONTROL ROOM

Startup Test (SUT-28)

Test Objective

To demonstrate that the reactor can be brought from a normal initial steady-state power level down to the point where cooldown is initiated and is under control with reactor vessel pressure and water level controlled from outside the main control room.

Prerequisites

The appropriate preoperational tests have been completed; the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

The test will be performed at a low power level and will consist of demonstrating the capability to scram and initiate controlled cooling from outside the control room. The reactor is scrammed and isolated from outside the control room after a simulated control room evacuation. Reactor pressure and water level will be controlled using SRVs or RCIC. ~~and RHR from outside the control room during the subsequent cooldown~~. The cooldown will continue until RHR shutdown cooling mode is placed in service from outside the control room with cooling water supplied from the ultimate heat sink. Alternatively, verification of satisfactory operation of the RHR shutdown cooling mode from outside the control room may be done at some other, more convenient time during the startup program. Coolant temperature must be lowered at least 50°F while in the shutdown cooling mode at a rate that would not exceed the limits of the Technical Specifications. All other operator actions not directly related to vessel water level and pressure will be performed in the main control room. The plant will be maintained in hot standby condition for at least 30 minutes during the performance of this test.

TABLE 14.2-232 (Cont)

The following tests are performed:

Action	Test Conditions
12 1. Functionally check use of remote shutdown panels (RSP) to shutdown reactor.	a. Steady-state power operation at TC-2. b. Reactor initially critical with MSIVs open. c. T-G online.
2. Functionally check use of RSP to cooldown reactor.	
3. Functionally check use of RSP to place shutdown cooling systems in operation.	

Acceptance Criteria

Level 1:

Not applicable.

Level 2:

During a simulated control room evacuation, the reactor must
12 be brought to the point where cooldown is initiated and
under control, and the reactor vessel pressure and water
level are controlled using equipment and controls outside
12 the control room.

TABLE 14.2-233

RECIRCULATION FLOW CONTROL
VALVE POSITION CONTROL

Startup Test (SUT-29A)

Test Objective

To demonstrate the recirculation flow control system's capability, while in the valve position (POS) mode. 22

Prerequisites

The appropriate preoperational tests have been completed; the SORC has reviewed and approved the test procedures and initiation of testing. All controls are checked and instrumentation calibrated.

Test Procedure

The testing of the recirculation flow control system follows a building-block approach while the plant is ascending from low to high power levels. Components and inner control loops are tested first, followed by drive flow control and plant power maneuvers to adjust and then demonstrate the outer loop controller performance. Preliminary component and valve position loop tests are run when the plant is in cold shutdown in order to visually observe the hydraulic cylinder response. While operating at low power with the pumps using the low-frequency power supply, small step changes are input into the position controller and the responses recorded. 22

The following test is performed:

Action	Test Conditions
1. Small and large step changes input into position controller.	a. Prior to plant heatup, reactor shutdown, recirc pumps off. (Preoperational testing results may be used to satisfy this testing requirement.)

2. Small step changes input into position controller.	a. Before or at TC-1 with pumps using low frequency power supply; at TC-3; ~~between TC 5~~ and 6. b. Recirculation system in POS mode; other systems in NORM mode.

Acceptance Criteria

Level 1:

The transient response of any recirculation system-related variables to any test input must not diverge.

Level 2:

1. Recirculation system-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

2. Maximum rate of change of valve position shall be 10 ± 1 percent/sec.

   During TC-3 and TC-6 while operating on the high speed (60 Hz) source, gains and limiters shall be set to obtain the following response.

3. Delay time for position demand step shall be:

   For step inputs of 0.5 percent to 5 percent — ≤0.15 sec

   22 For step inputs of 0.2 percent to 0.5 percent — (see Figure 14.2-233-1)

4. Response time for position demand step shall be:

   For step inputs of 0.5 percent to 5 percent — ≤0.45 sec

   22 For step inputs of 0.2 percent to 0.5 percent — (see Figure 14.2-233-1)

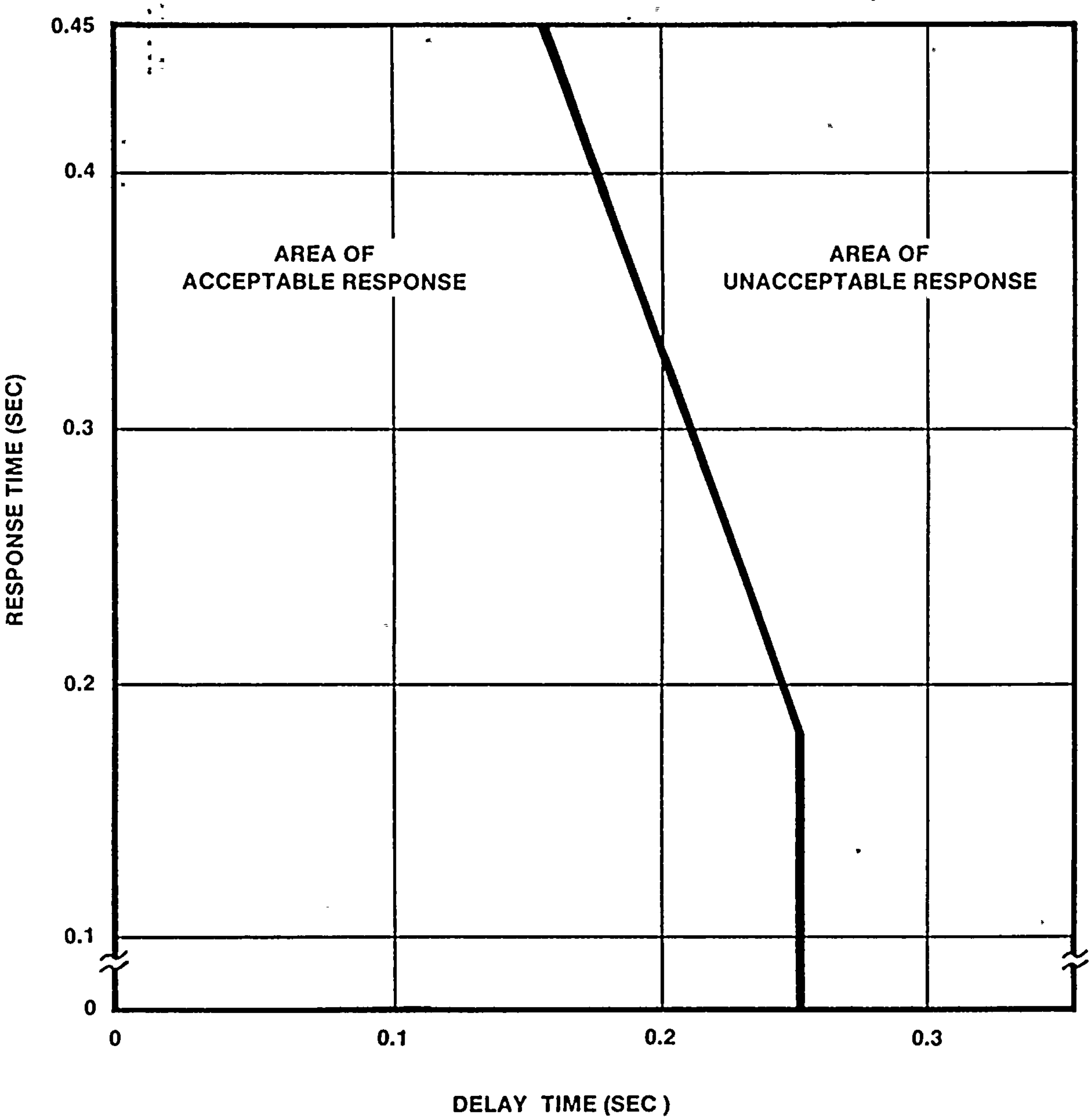


FIGURE 14.2-233-1

TRADEOFF CURVE FOR STEP SIZES 0.2% TO 0.3%

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
FINAL SAFETY ANALYSIS REPORT

TABLE 14.2-234 (Cont)

4. The response time for flow demand step (≤5 percent) must be 1.1 sec or less.

5. The maximum allowable flow overshoot for step demand of ≤5 percent of rated must be 6 percent of the demand step.

6. The flow demand step settling time must be ≤6 sec. 16

Flux Loop Criteria

Level 1:

The flux loop response to test inputs must not diverge.

Level 2:

1. Flux overshoot to a flux demand step must not exceed 2 percent of rated for a step demand of ≤20 percent of rated.

2. The delay time for flux response to a flux demand step must be ≤0.8 sec.

3. The response time for flux demand step must be ≤2.5 sec.

4. The flux setting time must be ≤15 sec for a flux demand step ≤20 percent of rated.

5. The flow control system shall be adjusted to limit the maximum core flow to 102.5% of rated by limiting the flow control valve opening position.

TABLE 14.2-234 (Cont)

For immediate commercial operation, the flux loop is set
22 slower, and the operator limits his action in the manual mode. If PCIOMRs are ever withdrawn, the tested faster auto settings can be inserted onto the controller with only a brief dynamic test, rather than a full startup test.

The following tests are performed:

Action	Test Conditions
1. Large and small step and ramp inputs.	a. ~~Between TC-2 and 3.~~ At TC3 b. Recirculation system in FLO and FLX modes; other systems in NORM mode. 22 c. Normal power sources to be used as applicable.
2. Step ~~and ramp input~~ changes to demonstrate satisfactory response.	a. ~~Between TC-5 and 6.~~ At TC6 b. Recirculation system in FLO and FLX modes; other systems in NORM mode.

Acceptance Criteria

Flow Loops Criteria

Level 1:

The transient response if any recirculation system-related variable to any test input must not diverge.

Level 2:

1. The decay ratio of the flow loop response to any test inputs must be <0.25.

2. The flow loops provide equal flows in the two loops during steady-state operation. Flow loop gains should be set to correct a flow imbalance in about 20 ± 5 sec.

3. The delay time for flow demand step (≤5 percent) must be 0.4 sec or less.

TABLE 14.2-234 (Cont)

Flux Estimator Test Criteria

Level 1:

Not applicable.

Level 2:

1. Switching between estimated and sensed flux should not exceed 5 times/5 min at steady state.

2. During flux step transient there should be no switching to sensed flux or if switching does occur, it should switch back to estimated flux within 20 sec of the start of the transient.

Flow Control Valve Duty Test Criteria

Level 1:

Not applicable.

Level 2:

The flow control valve duty cycle in any operating mode must not exceed 0.2 percent - Hz. Flow control valve duty cycle is defined as:

$$\frac{\text{Integrated valve movement in percent (\% Hz)}}{\text{2 x span (in sec)}}$$ 22

TABLE 14.2-236

RECIRCULATION SYSTEM, TWO-PUMP TRIP

Startup Test (SUT-30B)

Test Objective

To record and verify acceptable performance of the recirculation two-pump circuit trip system.

Prerequisites

The appropriate preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

In case of higher power turbine or generator trips, there is an automatic opening of circuit breakers in the pump power supply. The result is a fast core flow coastdown that helps reduce peak neutron and heat flux in such events. This two-pump ~~trip~~ transient test verifies that this flow coastdown is satisfactory prior to the high power turbine generator trip tests and subsequent operation (in TC 6). 22

The following test is performed:

Action	Test Conditions
Simulate TG-initiated RPT to trip all four RPT breakers simultaneously. (SUT-33, Drywell Piping Vibration, can be done in conjunction with this test.)	a. At TC-3 above 50 percent rated power and at 95 percent or more of rated core flow. ~~but before SUT-27, Turbine Trip and Generator Load Rejection.~~ 22 b. All systems in NORM mode. Water level may be lowered to avoid possible turbine trip scram.

TABLE 14.2-236 (Cont)

Acceptance Criteria

Level 1: excluding the first 0.25 seconds

The two-pump drive flow coastdown transient during the first 3 ~~8~~ sec ~~is~~ must be bounded by limiting curves. ~~These curves will be derived from time-delay data related to the data acquisition system utilized during the power ascention program. Once the data is obtained, the curves will be developed and added to the startup test specification.~~ These curves are site specific and will be supplied by GE-San Jose prior to startup.

Level 2:

~~This criteria is also from the bounding curves to be developed later, as stated above.~~

Not Applicable

(The limiting curves will be determined based upon measurement of the recirculation elbow flow meters transmitter time delay.)

TABLE 14.2-240

LOSS OF TURBINE GENERATOR AND OFFSITE POWER

Startup Test (SUT-31)

Test Objective

To determine the electrical equipment and reactor transient performance during the loss of auxiliary power.

Prerequisites

The appropriate preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

The loss of auxiliary power test is performed at 20 to 30 percent of rated power. The proper response of reactor plant equipment, automatic switching equipment, and the proper sequencing of the diesel generator load are checked. Appropriate reactor parameters are recorded during the resultant transient. The loss of power will be maintained long enough for plant conditions to stabilize (≥30 min). Systems which do not affect vessel level and pressure may be manually started and operated, as necessary.

The following test is performed:

Action	Test Conditions
After transferring auxiliary loads to the unit auxiliary transformer and starting main turbine dc oil pump; ~~use the trip relay to~~ trip 77 the main generator. (SUT-33 and 5 ~~Action Item 1,~~ can be done in conjunction with this test.)	a. At TC-2. b. Recirculation system in POS mode. All other systems in NORM mode. 22

TABLE 14.2-240 (Cont)

Acceptance Criteria

Level 1:

1. All safety systems such as the RPS, diesel generators, and HPCS must function properly without manual assistance, and HPCS and/or RCIC system action, if necessary, shall keep the reactor water level above the initiation level of the LPCS, LPCI, ADS, and MSIV closure. Diesel generators shall start automatically.

22 2.

Level 2:

1. Proper instrument display to the reactor operator shall be demonstrated, including power monitors, pressure, water level, control rod position, suppression pool temperature, and reactor cooling system status. Displays shall not be dependent on specially installed instrumentation.

22 2. If safety/relief valves open, the temperature measured by thermocouples on the discharge side of the safety/relief valves must return to within 10°F of the temperature recorded before the valve was opened.

2. Bypass flow should be ≥ 80% of bypass valve system's current capacity (based on % power at time of trip) within 0.3 seconds of start of stop/control valve closure.

TABLE 14.2-241

DRYWELL PIPING VIBRATION

Startup Test (SUT-33)

Test Objectives

1. To verify that the vibration of the reactor recirculation is within acceptable limits. 22

2. To verify that stresses are within code limits during operating transient loads.

Prerequisites

The appropriate preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

This test is an extension of the system expansion test (SUT-17). Consult the specification of SUT-17 for piping considered to be within the scope of testing. 22

Because of limited access due to high radiation levels, no visual observation is required during the startup phase of the testing. Remote measurements of piping vibrations are made during the following steady-state conditions:

1. Recirculation at minimum flow. ~~and coincident temperature.~~ 22

TABLE 14.2-241 (Cont)

2. Recirculation at 50 percent ±5 percent of rated flow and operating temperature.

22 3. Recirculation at 75 percent ±5 percent of rated flow and operating temperature.

22 4. Recirculation at 100 percent of rated flow.

22 During the operating transient load testing the amplitude of the piping displacement ~~and number of cycles per transient of the recirculation piping~~ ~~are~~ is measured, and the displacements compared with acceptance criteria. Remote ~~vibration and~~ deflection measurements are taken during the following transients:

1. Recirculation pump start.

2. Recirculation pump trip at 100 percent of rated flow.

22 The locations to be monitored and predicted displacements for the monitored locations in each plant will be provided ~~later~~ by the General Electric Piping Response Measurement Data Sheet 22A5405 4V. The following tests are performed:

Action	Test Conditions
Record recirculation loop vibration	a. Recirc. at minimum flow at TC-1 b. At 50, 75, and at approximately 100% of rated recirculation flow on 100% load line. c. In conjunction with recirculation pumps starts and transfers ~~trips~~ (Tests 30A and B) at TC-3 and 6. d. In conjunction with Test 71 startup, shutdown and while at 100% of rated RHR flow in the shutdown cooling mode.

22

TABLE 14.2-242 8

RECIRCULATION SYSTEM FLOW CALIBRATION 8

Startup Test (SUT-35)

Test Objective

To perform complete calibration of the installed recirculation system flow instrumentation.

Prerequisites

The appropriate preoperational tests have been completed and the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

During the testing program at operating conditions that allow the recirculation system to be operated at rated flow at power, the jet pump flow instrumentation is adjusted to provide correct flow indication based on jet pump flow. After the relationship between drive flow and core flow is established, the flow biased APRM/RBM (rod block monitor) system is adjusted to match this relationship.

The following test is performed:

Action	Test Conditions
Take recirculation system data and recalibrate instrumentation.	a. At TC-3. b. At TC-6.

Acceptance Criteria

Level 1:

Not applicable.

Level 2:

1. Jet pump flow instrumentation is adjusted in such a way that the jet pump total flow recorder provides a correct core flow indication at rated conditions.

TABLE 14.2-242 (Cont)

2. The APRM/RBM flow-bias instrumentation is adjusted to function properly at rated conditions.

~~3. The flow control system shall be adjusted to limit the~~
12 ~~maximum core flow to 102.5 percent of rated by limiting~~
~~the flow control valve opening position.~~

3. The calculated jet pump M-ratio shall not be < 0.2 points below prediction.

4. The nozzle and riser plugging criteria shall not be exceeded.

TABLE 14.2-243

REACTOR WATER CLEANUP SYSTEM

Startup Test (SUT-70)

Test Objective

To demonstrate specific aspects of the mechanical ability of the RWCU. (This test, performed at rated reactor pressure and temperature, is actually the completion of the preoperational testing that could not be done without nuclear heating.)

Prerequisites

The preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

With the reactor at rated temperature and pressure, process variables are recorded during steady-state operation in the following ~~three~~ modes, as defined by the system process diagram: hot standby, and normal, ~~and blowdown~~. A comparison of the bottom head flow indicator and the RWCU inlet flow indicator is made during these modes. The RWCU system sample station is tested at hot process conditions as part of SUT 1. 22

The following test is performed:

Action	Test Conditions
Take heat balance and pressure data.	a. Reactor at rated temperature and pressure during heatup.
	b. Cleanup system operate in hot standby and normal~~, and blowdown modes.~~

Acceptance Criteria

Level 1:

Not applicable.

TABLE 14.2-243 (Cont)

Level 2:

1. The temperature at the tube side of the nonregenerative heat exchangers does not exceed 130°F in the ~~blowdown~~ hot standby mode or 120°F in the normal mode.

2. The pump available NPSH at least 13 ft during the hot standby mode is as defined in the process diagrams. *

3. The cooling water supplied to the nonregenerative heat exchangers shall be less than 6 percent above the flow corresponding to the heat exchanger capacity (as determined from the process diagram) and the existing temperature differential across the heat exchangers. The outlet temperature shall not exceed 180°F.

22 4. Recalibrate bottom head flow indicator against RWCU flow indicator if the deviation is greater than 25 gpm.

22 5. Pump vibration shall be less than or equal to 2 mils peak-to-peak (in any direction) as measured on the bearing housing, and 2 mils peak-to-peak shaft vibration as measured on the coupling end.

* If measurements and calculations made during the system preop test show that NPSH requirements for this mode can be met then this requirement need not be addressed during startup testing.

TABLE 14.2-244

RESIDUAL HEAT REMOVAL SYSTEM

Startup Test (SUT-71)

Test Objective

To demonstrate the ability of the RHR system to:

1. Remove heat from the reactor system so that the refueling and nuclear system servicing can be performed.

2. Condense steam while the reactor is isolated from the main condenser.

Prerequisites

The appropriate preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

With the reactor at a convenient thermal power, the steam condensing mode of the RHR system is tuned and demonstrated. Condensing heat exchanger performance characteristics are demonstrated. Final demonstration of the condensing mode is done from an isolated condition. During the first suitable reactor cooldown, the shutdown cooling mode of the RHR system is demonstrated. Unfortunately, the decay heat load is insignificant during the startup test period. Use of this mode with low core exposure could result in exceeding the 100°F/hr cooldown rate of the vessel if both RHR heat exchangers are used simultaneously. Late in the test program after accumulating significant core exposure, this demonstration would more adequately demonstrate the heat exchanger capacity. The RHR heat exchangers will also be tested in the suppression pool cooling mode. | 22

The following tests are performed:

or during the first shutdown after the test program

TABLE 14.2-244 (Cont)

Action	Test Conditions
1. Controller adjustment based on subsystem perturbations	a. Reactor not isolated above 10% rated power but ≤25% rated power. b. RHR system in steam condensing mode. c. RCIC flow to CST.
2. Demonstration of steam condensing mode.	a. Reactor at hot standby and isolated. b. RCIC flow to RPV.
3. Take heat exchanger capacity data.	a. RHR in shutdown cooling mode. b. After trip or cooldown from TC-6 in order to provide sufficient decay heat. c. RHR in suppression pool cooling mode.

or during the first shutdown after the test program

Acceptance Criteria

Level 1:

The transient response of any system-related variable to any test input must not diverge.

Level 2:

or both

1. The RHR system must be capable of operating in the steam condensing, suppression pool, cooling, and shutdown cooling modes (with both one ~~and two~~ heat exchangers) at ~~the flow rates and temperature differentials~~ indicated on the process diagrams.

22 2. System-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

heat removal rates equivalent to or greater than the values

TABLE 14.2-245

OFF-GAS SYSTEM

Startup Test (SUT-74)

Test Objective

The purpose of this test is to verify the proper operation of the off-gas system over its expected operating range.

Prerequisites

The appropriate preoperational tests have been completed. The SORC has reviewed and approved the test procedure and the initiation of testing. Instrumentation has been checked and calibrated as appropriate. The following systems must be operable to the extent required to support testing:

- Condenser air removal.
- Auxiliary steam.
- Turbine building closed loop cooling. 22
- Service air.
- Radiation monitoring.

Test Procedure

The following off-gas system tests will be conducted at various power levels throughout plant startup while at steady-state conditions:

1. Hydrogen analyzer - Check that the hydrogen analyzer is functioning and record the level of hydrogen in the recombiner effluent.

2. Dewpoint - Check that the dewpoint in the off-gas system complies with design temperatures.

3. Temperature - Monitor the temperature of the charcoal absorbers, active and standby catalytic recombiner, and freeze-out dryer discharge to see that the specified limits are met.

4. Recombiner performance - As the recombiner performance is least efficient in the lower power

range; it will be inspected closely in this range for correct initial operation.

5. Recombiner feed - Readings of the off-gas flow are taken to ensure that a hydrogen concentration of less than or equal to 4 percent is maintained in the recombiner feed.

6. Radionuclide residence times - Provided that reasonable and sufficient fission gasses are present in the off-gas, measurements will be made of at least one radionuclide to determine the decontamination factor(s) across one or several charcoal beds.

22 7. HEPA filters - If sufficient particulate fission gas daughter products are present, measurements of decontamination factors across the filters will be made. This is to confirm that the filters are operating properly during normal operating conditions.

8. Radiolytic gas production - Calculate the radiolytic gas production rate based on recombiner differential temperatures and verify that the production rate is within the design value.

9. Freeze-out dryer performance - Monitor the effluent dewpoint of the freeze-out dryer during its operating cycles to verify that discharge limits are met.

Insert 10 (SEE NEXT PAGE)

Acceptance Criteria

Level 1:

The release of radioactive gaseous and particulate effluents must not exceed the limits specified in the site technical specifications.

Level 2:

1. The system flow, pressure, temperature, and dewpoint shall comply with the process data sheets supplied to the site.

10. The test data will then be provided to the appropriate Engineering personnel for evaluation to verify the system will perform adequately under design conditions.

TABLE 14.2-245 (Cont)

2. The catalytic recombiner, hydrogen analyzer, freeze-out dryers, activated carbon beds, and filters shall be working properly during operation, i.e., there shall be no gross malfunctioning of these components. 22

TABLE 14.2-301

DRYWELL COOLING SYSTEM

Startup Test (SUT-75)

Test Objective

To demonstrate the capability of the drywell cooling system to maintain peak and average drywell temperatures within the maximum design limits during power operation at rated temperature and pressure.

Prerequisites

The appropriate preoperational tests have been completed. The SORC has reviewed and approved the test procedures and the initiation of testing. Instrumentation has been checked and calibrated as appropriate. The service water and closed loop cooling systems are operational to the extent required to conduct the test.

Test Procedure

The following data will be recorded and evaluated at the test conditions listed. 22

Action	Test Conditions
1. Record temperature and flow data to perform a heat balance across the coolers, check average space temperature, and check suspected hot spot temperatures.	a. ~~After~~ During heatup to rated temperature and pressure, TC2 and TC6
2. Check suspected hot spot temperatures as well as average space temperature during both normal and post-scram conditions.	a. TC-~~3~~ 2 and 6

Acceptance Criteria

Level 1:

Drywell average air space temperature shall not exceed the limit specified in plant technical specifications.

Level 2:

The maximum temperature measured in any area of the Drywell shall not exceed the design limits specified in Table 9.4-1.

TABLE 14.2-302

ESF AREA COOLING

Startup Test (SUT-76)

Test Objective

The purpose of this test is to verify Aux. Bldg MCC, that the unit coolers serving the RCIC, RHR, LPCS, HPCS, SGTS, Service water, bay and Standby Diesel Generator ~~equipment~~ Control rooms can maintain the equipment room temperature below the maximum design limits under postulated accident conditions.

Prerequisites

The appropriate preoperational tests have been completed. The SORC has reviewed and approved the test procedures and the initiation of testing. Instrumentation has been checked and calibrated as appropriate. The service water system is operational to the extent required to conduct the test.

Test Procedure 22

The ESF areas listed above will be isolated from the normal ventilation system and major equipment in the area ~~will be~~ run in the mode providing the maximum practical heat load. Numerous temperature measurements will be made in the area. Adequate temperature and flow data will be collected to perform a heat balance across the area coolers under test conditions. The test data will then be provided to appropriate engineering personnel for evaluation to verify the system will perform adequately under design basis conditions.

Acceptance Criteria

Level 1:

All ESF area air space temperatures measured shall not exceed the design limits specified in Table 9.4-1.

Level 2:

Evaluation of test data shall demonstrate that all ESF area air space temperatures will remain below the design limits in Table 9.4-1 under design basis conditions.

TABLE 14.2-303

BOP PIPING VIBRATION

Startup Test (SUT-77)

Test Objective

1. To verify that steady-state and/or transient piping vibration for the main steam (including relief valve discharge), residual heat removal, feedwater, reactor core isolation cooling, and condensate systems are within acceptable limits.

2. To verify that steady-state vibrations for small bore piping and essential instrumentation lines on main steam, nuclear steam supply, feedwater, reactor plant sampling, residual heat removal, and reactor core isolation cooling are within acceptable limits.

Prerequisites

The appropriate preoperational tests and generic pipe vibration tests have been completed, and the SORC has reviewed and approved the test procedures and the initiation of testing. Instrumentation has been checked and calibrated as appropriate.

Test Procedure

Due to the high radiation levels involved no visual observations are performed in this test. Remote monitoring of piping vibration will be utilized. The locations to be monitored and the corresponding predicted displacements will be provided in the startup test procedure.

The following tests are performed:

Action	Test Conditions
~~1. Record vibration of RCIC and main steam lines.~~	~~a. At 25, 50, 75, and approximately 100% of rated main steam flow.~~
	~~b. In conjunction with turbine generator trip (SUT-27 at TC-2,6) and relief valve capacity checks (SUT-26) between TC-2 and 6.~~ 24

See following pages

Action	Test Conditions
1. Record vibration of RCIC lines	a. In conjunction with RCIC pump start, rated pump flow, and RCIC pump trip (SUT-14 at Hu and 1)
2. Record vibration of main steam lines	a. At approximately 25, 50, 75 and 100% of rated thermal power
	b. In conjunction with pressure controller setpoint changes (SUT-22 at TC-2 and 4), MISV closure (SUT-25 at TC-3 and 5), relief valve capacity checks (SUT-26 at TC-1) and turbine generator trip (SUT-27 at TC-2)
3. Record vibration of main steam relief valve lines	a. In conjunction with relief valve capacity checks (SUT-26 at TC-1)
4. Record vibration of feedwater and condensate lines	a. At approximately 25, 50, 75 and 100% of rated thermal power
	b. In conjunction with turbine generator trip (SUT-27 at TC-2 and 6) and feedwater system tests (SUT-23 at TC-1, 2, 5 and 6)
5. Record vibration of residual heat removal lines	a. At approximately 25, 50, 75 and 100% of rated thermal power
	b. In conjunction with RHR steam condensing mode and shutdown cooling mode (SUT-71 at TC-1 and 6)
6. Record vibration of reactor plant sampling lines	a. In conjunction with feedwater system test (SUT-23 at TC-6)
7. Record vibration of recirculation system instrumentation	a. In conjunction with rated recirculation flow on 100% load line
8. Record vibration of reactor vessel level indicator instrumentation (nuclear boiler instrumentation, ISC) lines	a. At approximately 25, 50, 75 and 100% of rated thermal power
9. Record vibration of main steam instrumentation lines	a. In conjunction with MSIV closure (SUT-25 at TC-6)

TABLE 14.2-304

BOP SYSTEM EXPANSION

Startup Test (SUT-78)

Test Objective

To verify that BOP piping systems are free to expand and move without unplanned obstruction or restraint during system heatup and cooldown cycles and to verify that the associated measured displacements are within specified limits.

Prerequisites

The appropriate preoperational tests have been completed and the SORC has reviewed and approved the test procedures and the initiation of testing. Instrumentation has been checked and calibrated as appropriate.

Test Procedure

Visual inspections will be performed, to the extent possible, to verify freedom of movement. In addition, scribers and remote sensors will be utilized to obtain displacement readings. 22

Action	Test Conditions
1. Visual inspection for: reactor water cleanup, high pressure core spray, low pressure core spray, residual heat removal, reactor core isolation cooling, feedwater, and condensate systems.	a. Prior to initial heatup at ambient conditions. b. At the reactor vessel temperature plateau of 300 ± 50°F. c. At the reactor vessel temperature plateau of 500 ± 50°F rated. d. At the end of the first heatup/cooldown cycle (near ambient conditions).

TABLE 14.2-304 (Cont)

2. Record remote sensor displacement readings for: reactor water cleanup, high pressure core spray, low pressure core spray, residual heat removal, control rod drive, reactor core isolation cooling, main steam, ~~main steam safety relief~~ and feedwater systems.
   a. Prior to initial heatup at ambient conditions.
   b. At the reactor vessel temperature plateau of 300 ± 50°F.
   c. At the reactor vessel temperature plateau of 500 ± 50°F rated.
   d. At the end of the first heatup/cooldown cycle (near ambient conditions).

3. Record scriber displacement readings for: reactor water cleanup, reactor core isolation cooling, feedwater, main steam, and condensate systems.
   a. Prior to initial heatup at ambient conditions.
   b. At the reactor vessel temperature plateau of 300 ± 50°F.
   c. At the reactor vessel temperature plateau of 500 ± 50°F rated.
   d. At the end of first cycle heatup (near ambient conditions).

4. Record remote sensor displacement readings for Main Steam Safety Relief
   a. During SRV Testing at TC-1

~~4~~ 5. Record remote sensor displacements for feedwater system.
   a. Upon feedwater system obtaining within ±20°F of its rated temperature during TC-6.

<u>Acceptance Criteria</u>

<u>Level 1:</u>

1. There are no obstructions which will interfere with the thermal expansion of the above piping systems.

TABLE 14.2-304 (Cont)

2. The displacements at the established transducer and scriber locations shall not exceed the allowable values provided in the BOP thermal expansion procedure. The allowable values of displacement shall be based on not exceeding ASME Section III code stress allowables. 22

Level 2:

1. The displacements at established transducer and scriber locations shall not exceed the expected values as provided in the BOP system expansion procedure.

Add this New Abstract

Add This New Abstract

Table 14.2-307

Drywell High Energy Penetrations

Startup Test (SUT-81)

Test Objective:

The purpose of this test is to demonstrate the capability of the Drywell High Energy Penetrations to maintain the surrounding concrete below design temperature limits.

Prerequisites:

The SORC has reviewed and approved the test procedure and the initiating of testing. Instrumentation has been checked and calibrated as appropriate.

Test procedure:

The following Drywell Penetration System test will be performed at various power levels during plant startup while at steady-state conditions:

1. Temperature - Monitor the thermal rise of the process piping, flued head and the liner insert junction. Monitor the wall insert annular space if practical.

2. The test data will then be provided to the appropriate engineering personnel for evaluation to verify the system will perform adequately under design and accident conditions.

Acceptance Criteria:

Level 1:

NONE

Level 2:

1. The temperature of the concrete surrounding the high energy Drywell Penetrations shall not exceed 200 degrees Fahrenheit during normal operation.

2. Evaluation of test data shall demonstrate that the temperature of the concrete surrounding the high energy Drywell Penetrations will not exceed 340 degrees Fahrenheit during accident conditions.

TABLE 14.2-403

QUALIFICATION OF GE PRINCIPAL TESTING PERSONNEL DURING STARTUP TESTING | 22

The GE Site Operations Manager meets the equivalent of ANSI N45.2.6, 1978, discussed for a Level III person. The Operations Manager is normally present for preoperational testing and will be SRO certified under the GE certification program.

The GE Operations Superintendent meets the equivalent of ANSI N45.2.6, 1978, discussed for a Level III person or a Level II person. The Operations Superintendent is normally present for preoperational testing and will be SRO certified under the GE certification program.

The GE ~~Shift Superintendents~~ STO Engineers meet the equivalent of ANSI N45.2.6, 1978, discussed for a Level II person. They will also be SRO certified under the GE certification program.

The GE Lead Startup Test Design and Analysis Engineer meets the equivalent of ANSI N45.2.6, 1978, discussed for a Level III person or a Level II person. He is qualified at the time of appointment to the position.

The GE Startup Test Design and Analysis Engineers meet the equivalent of ANSI N45.2.6, 1978, discussed for a Level II person.

The GE Startup Control and Instrumentation Engineers meet the equivalent of ANSI N45.2.6, 1978, discussed for a Level II person.

The GE Startup Chemist meets the equivalent of ANSI N45.2.6, 1978, discussed for a Level II person.

In addition, all GE personnel listed above will meet ANSI 3.1, 1978, Section 4.3.2 minimum qualifications.

ATTACHMENT 1

ADDITIONAL RELIEF REQUESTS

Nine Mile Point Unit 2

Relief Request No: RR-IWB-1

1. Identification of Components

Pages 3 through 4 of 4 identifies those CRD housing welds for which relief from partial or total ASME XI exam is required.

2. ASME Section XI Requirements

Volumetric or Surface examinations are required for CRD housing welds in accordance with table IWB2500-1, Category B-O, Item no. B14.10.

3. Basis for Relief

The examinations of the CRD housing welds as identified in the NMP2 PSI Program Plan can only be performed on a limited scope due to the inherent obstructions caused by the surrounding cables, tubing and foundations which are integral parts of the CRD assembly. The approximate extent of the best effort surface and volumetric ASME XI preservice exams and the limiting permanent obstructions are indicated on pages 3 through 4 of 4. It is not practical to remove or replace these obstructions due to the congestion in the CRD assembly area. The integrity of the CRD housings has also been previously verified by non destructive examination during fabrication and erection under ASME Sect. III.

4. Inspection Period for Relief Request

Pre Service Inspection

5. Alternate Tests or Examinations

ASME Sect. XI volumetric and surface exams are performed. ASME Sect. III liquid penetrant and radiography exam results will also be used.

6. Schedule for Implementing Alternate Test

During Pre Service Inspection

7. **Impact to Overall Plant Level of Quality**

   Overall Plant Quality is not Impacted

8. **Preservice Examination Results**

   ASME XI volumetric and surface exam results and ASME III liquid penetrant and radiography exam results will be submitted in the Final Summary Report.

9. **Radiation Considerations**

   None

Nine Mile Point Unit 2

Relief Request No: RR-IWB-1

Weld	Extent of Exam	Obstructions
RPV-CRDH-ØØ1A	17Ø°	Cables, Tubing
ØØ1B	27Ø°	Foundation
ØØ2A	9Ø°	Cables, Tubing
ØØ2B	27Ø°	Foundation
ØØ3A	17Ø°	Cables, Tubing
ØØ3B	27Ø°	Foundation
ØØ4A	17Ø°	Cables, Tubing
ØØ4B	27Ø°	Foundation
ØØ5A	17Ø°	Cables, Tubing
ØØ5B	27Ø°	Foundation
ØØ6A	17Ø°	Cables, Tubing
ØØ6B	27Ø°	Foundation
ØØ7A	17Ø°	Cables, Tubing
ØØ7B	27Ø°	Foundation
ØØ8A	Inaccessible	Foundation,Tubing
ØØ8B	Inaccessible	Foundation,Tubing
ØØ9A	Inaccessible	Foundation,Tubing
ØØ9B	Inaccessible	Foundation,Tubing
Ø1ØA	Inaccessible	Foundation,Tubing
Ø1ØB	Inaccessible	Foundation,Tubing
Ø11A	Inaccessible	Foundation,Tubing
Ø11B	Inaccessible	Foundation,Tubing
Ø12A	Inaccessible	Foundation,Tubing
Ø12B	Inaccessible	Foundation,Tubing
Ø13A	17Ø°	Tubing
Ø13B	Inaccessible	Foundation,Tubing
Ø14A	17Ø°	Tubing
Ø14B	Inaccessible	Foundation,Tubing
Ø15A	17Ø°	Tubing
Ø15B	Inaccessible	Foundation,Tubing
Ø16A	17Ø°	Tubing
Ø16B	Inaccessible	Foundation,Tubing
Ø17A	17Ø°	Tubing
Ø17B	Inaccessible	Foundation,Tubing
Ø18A	17Ø°	Tubing
Ø18B	Inaccessible	Foundation,Tubing
Ø19A	17Ø°	Tubing
Ø19B	Inaccessible	Foundation,Tubing
Ø2ØA	17Ø°	Tubing
Ø2ØB	Inaccessible	Foundation,Tubing
Ø21A	17Ø°	Tubing
Ø21B	Inaccessible	Foundation,Tubing
Ø22A	17Ø°	Tubing
Ø22B	Inaccessible	Foundation,Tubing
Ø23A	17Ø°	Tubing
Ø23B	Inaccessible	Foundation,Tubing

Nine Mile Point Unit 2

Relief Request No: RR-IWB-1

Weld	Extent of Exam	Obstructions
RPV-CRDH-Ø24A	17Ø °	Tubing
Ø24B	Inaccessible	Foundation,Tubing
Ø25A	17Ø °	Tubing
Ø25B	Inaccessible	Foundation,Tubing
Ø26A	17Ø °	Tubing
Ø26B	Inaccessible	Foundation,Tubing
Ø27A	Inaccessible	Foundation,Tubing
Ø27B	Inaccessible	Foundation,Tubing
Ø28A	Inaccessible	Foundation,Tubing
Ø28B	Inaccessible	Foundation,Tubing
Ø29A	Inaccessible	Foundation,Tubing
Ø29B	Inaccessible	Foundation,Tubing
Ø3ØA	Inaccessible	Foundation,Tubing
Ø3ØB	Inaccessible	Foundation,Tubing
Ø31A	Inaccessible	Foundation,Tubing
Ø31B	Inaccessible	Foundation,Tubing
Ø32A	Inaccessible	Foundation,Tubing
Ø32B	Inaccessible	Foundation,Tubing
Ø33A	Inaccessible	Foundation,Tubing
Ø33B	9Ø °	Foundation
Ø34A	Inaccessible	Cables
Ø34B	27Ø °	Foundation
Ø35A	1ØØ °	Cables, Tubing
Ø35B	27Ø °	Foundation
Ø36A	17Ø °	Cables, Tubing
Ø36B	27Ø °	Foundation
Ø37A	17Ø °	Cables, Tubing
Ø37B	27Ø °	Foundation
Ø38A	17Ø °	Cables, Tubing
Ø38B	27Ø °	Foundation
Ø39A	17Ø °	Cables, Tubing
Ø39B	9Ø °	Foundation
Ø4ØA	17Ø °	Cables, Tubing
Ø4ØB	9Ø °	Foundation

Nine Mile Point Unit 2

Relief Request No: RR-IWB-2

1. Identification of Components

Pg 3 of 8 identifies the RPV Nozzle to shell welds for which partial relief from ASME XI exam is required.

2. ASME Section XI Requirements

Volumetric examinations are required for RPV nozzle to shell welds in accordance with table IWB2500-1, Category BD, Item No. B3.90.

3. Basis for Relief

The automated examination of the RPV nozzle to shell welds as identified in the NMP2 PSI Program Plan can only be performed on a limited scope due to the nozzle to shell blend, vessel scanner tracks, other nozzles and mechanical limitations. The extent of the worst case limitations including description and sketches is shown on page 3 thru 8. The integrity of the subject welds has also been previously verified by non destructive examination during fabrication and erection under ASME Sect. III.

4. Inspection Period for Relief Request

Pre Service Inspection

5. Alternate Tests or Examinations

Ultrasonic Inspections performed in the vessel fabrication shop will also be used.

6. Schedule for Implementing Alternate Test

Previously performed in shop.

7. Impact to Overall Plant Level of Quality

Overall plant quality is not impacted.

8. Preservice Examination Results

ASME XI volumetric exam results and shop UT data will be submitted in the final report.

9. Radiation Considerations

None

NOZZLE	WELD AND/OR RASTER	EXTENT OF LIMITATION: ONE SIDED PERPENDICULAR DIRECTION	EXTENT OF LIMITATION: PARALLEL DIRECTION	LIMITATION CAUSED BY	TYPICAL SKETCH SHOWN ON PAGE
N1	KAØ1, KAØ2	Ø°-36Ø°, 19"-26"	Ø°-36Ø°, 19"-26"	Nozzle Blend	4
N2	KAØ3, thru KA12	Ø°-36Ø°, 1Ø"-2Ø" 215°-325°	Ø°-36Ø°, 1Ø"-2Ø" 215°-325°	1) Nozzle Blend 2) Vessel Scanner tracks - welds KAØ3,Ø4,Ø5,Ø6 Ø7,1Ø,12. 3) N9 Nozzle Welds KAØ5,1Ø,11	5
N4	KA17 thru KA22	Ø°-36Ø°, 11"-19"	Ø°-36Ø°, 11"-19"	Nozzle Blend	6
N5 N16	KA23, 32	Ø°-36Ø°, 11"-21" 215°-325°, 30°-140°	Ø°-36Ø°, 11"-21" 215°-325°, 30°-140°	1) Nozzle Blend 2) Vessel Scanner Tracks	7
N6	KA24, 25, 26	Ø°-36Ø°, 1Ø"-21"	Ø°-36Ø°, 1Ø"-21"	Nozzle Blend	8

EXAMINATION LIMITATION REPORT SHEET

Site/Unit NMP2

Page 3 of 5

Raster No. A01C1*

Package No. 1749C33

Weld No. RPV-KA01

Procedure No. 83A1749

*SEE A01C2 FOR COMPLETION of RASTER.

Rev. 0 F.C. No. 001,002

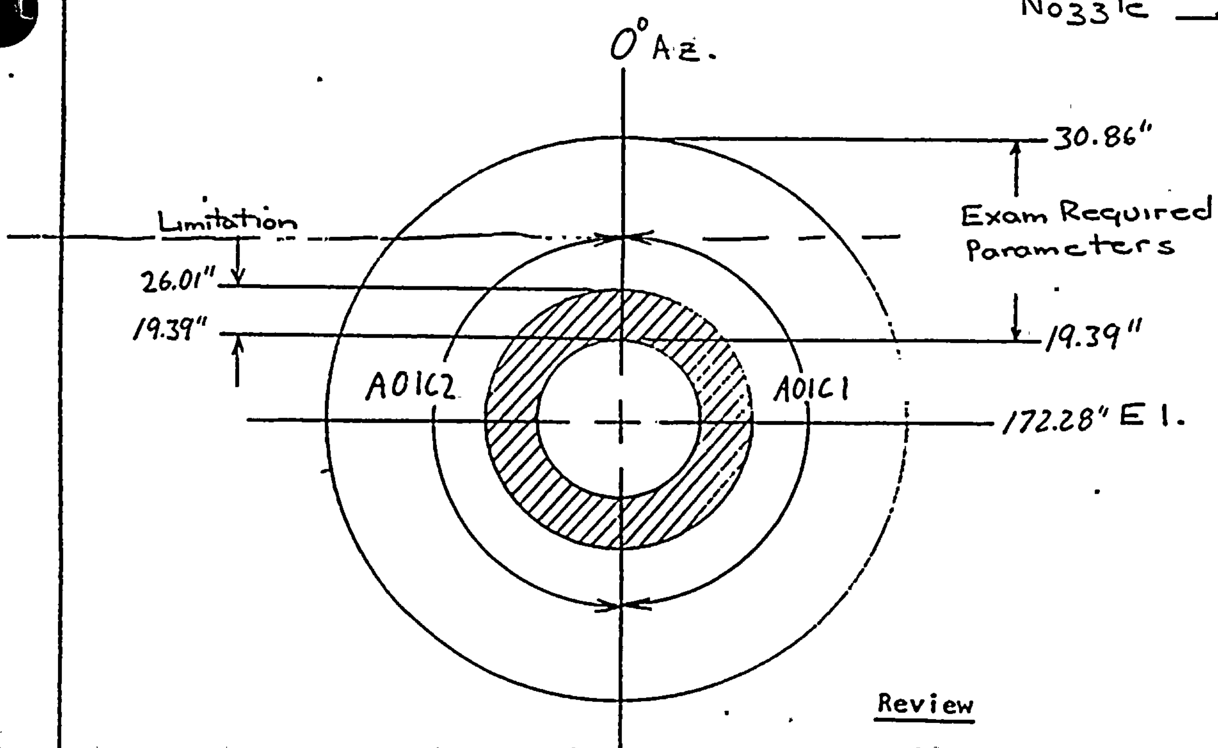
Limitation caused by NOZZLE BLEND

Area not scanned due to limitation

X-0° to 360° Y-19.39" to 26.01"

Comments (use sketch to provide clarity) REF DWG-NES 80E3536 REV 3

Nozzle N/A



Review

Date

Date

Date

Zam D. Dorgay LII 21 Feb 86

Prepared By [signature] Date 21FEB86

NES

EXAMINATION LIMITATION REPORT SHEET

Site/Unit NMP2 | Page 4 of 5

Raster No. A05I1 | Package No. 1749C18

Weld No. RPV-KA05 | Procedure No. 83A1749

Rev. 0 F.C. No. 001,002

Limitation caused by INSTRUMENTATION NOZZLE, ~~AND~~ TRACK #5 AND NOZZLE BLEND

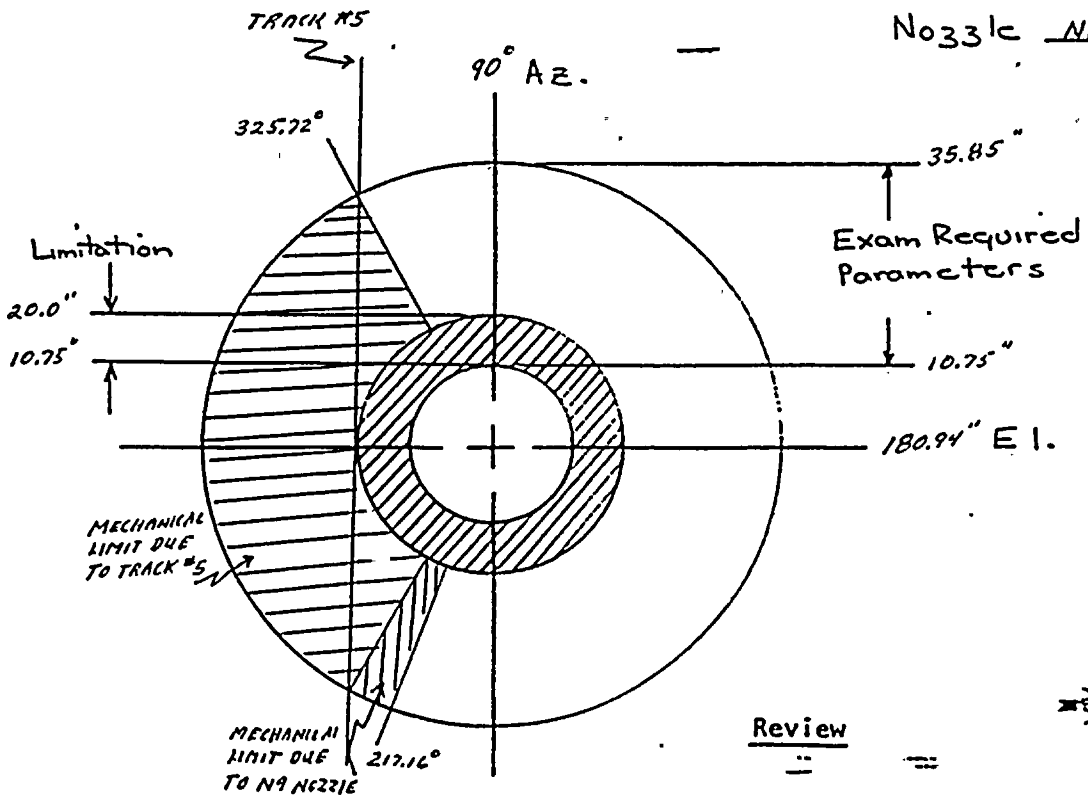
Area not scanned due to limitation

X – 0° to 360° Y – 10.75" to 20.0" NOZZLE BLEND,

X = 217.16° to 325.72° Y – 20.0" to 35.85" TRACK #5, AND NOZZLE

Comments (use sketch to provide clarity)

Nozzle N2C



Review

_______ Date _______

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_______ Date _______

Prepared By [signature] LII Date 2-19-86

Relief REQUEST NO. RR-IWB-2
PG 6 of 8

EXAMINATION LIMITATION REPORT SHEET

Site/Unit NMP2 | Page 2 of 5

Raster No. A17C1 | Package No. 1749C27

Weld No. RPV-KA17 | Procedure No. 83A1749

Rev. 0 | F.C. No. 001,002

Limitation caused by NOZZLE BLEND

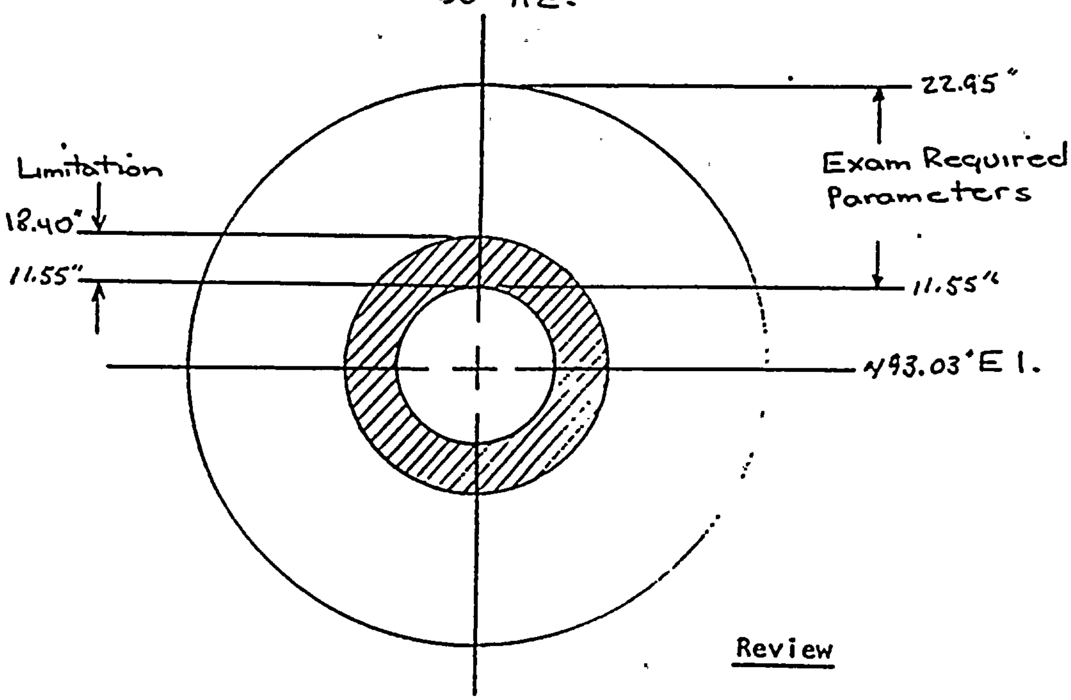
Area not scanned due to limitation

X - 0° - ~~361°~~ 360° Y - 18.40" -- 11.55"

Comments (use sketch to provide clarity)

REF DWG 80E3536 ~~REF~~ REV. 3

Nozzle N4A



Review

Date

Date

Date

Prepared By [signature] Date 2-17-86

RELIEF REQUEST NO. RR-IWB-2

PG 7 of 8

EXAMINATION LIMITATION REPORT SHEET

Site/Unit NMP2

Page 3 of 5

Raster No. A32C1

Package No. 1749C4

Weld No. KA32

Procedure No. 83A1749

Rev. 0 F.C. No. 001,002

Limitation caused by (1) Nozzle Blend and (2) Vessel Scanning tracks

Area not scanned due to limitation

X - 90° to 270° Y - 20.60" to 11.68" Nozzle Blend.

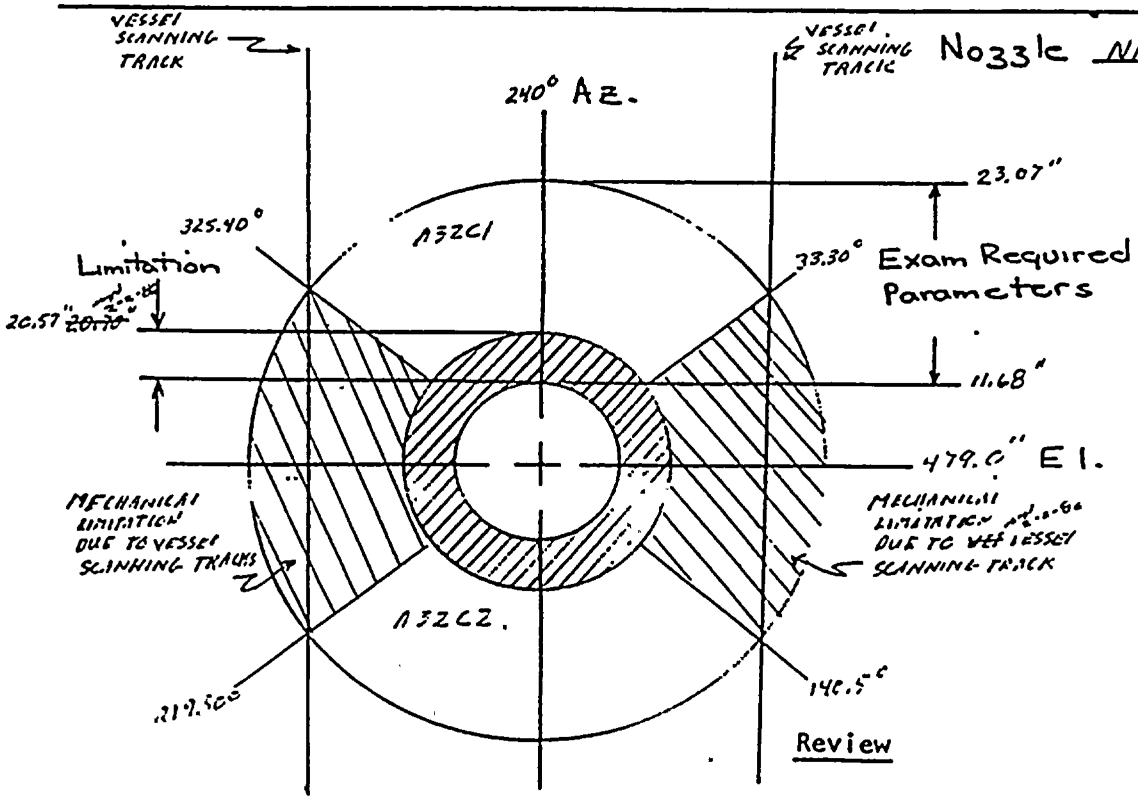
X - ~~0 to 360~~° 270° to 90° Y - 20.57" to 11.68" Nozzle Blend

X 33.30° to 140.5° Y = 20.57" to 23.07" Mechanical limit due to vessel scanning tra.

X - 219.50° to 325.40° Y = 20.57" to 23.07 Mechanical limit due to vessel scanning tra.

Comments (use sketch to provide clarity)

Nozzle N16A



Review

Date ______

Date ______

Date ______

Prepared By [signature] L-III Date 2-2-86.

nes

RELIEF REQUEST NO. RR-IWB-2
PG 80 of 8

EXAMINATION LIMITATION REPORT SHEET

Site/Unit NMP2

Raster No. A26I1

Weld No. RPV-KA26

Page 3 of 11

Package No. 1749C24

Procedure No. 83A1749

Rev. 0 F.C. No. 001,002

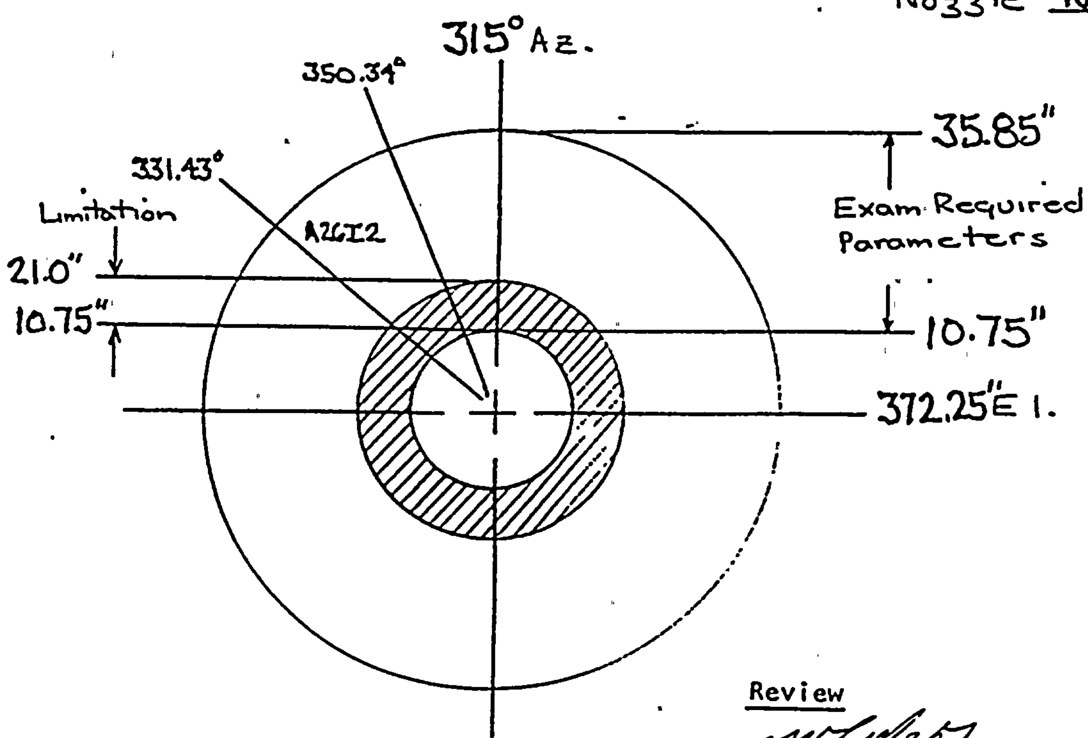
Limitation caused by NOZZLE BLEND

Area not scanned due to limitation

X - 0° TO 360° Y - 10.75" TO 21.0"

Comments (use sketch to provide clarity)

Nozzle N6C



Review

[signature] L-III Date 2-20-86

Date

Date

Prepared By [signature] Date 2-19-86

FORM NES 028-114/1 8/85 REF: Automated

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Nine Mile Point Unit 2

Relief Request No: RR-IWB-3

1. Identification of Components

Page 3 thru 5 identifies RPV seam welds for which partial relief from ASME XI exam is required.

2. ASME Section XI Requirements

Volumetric examinations are required for RPV seam welds in accordance with table IWB 2500-1, Category B-A, Item No's B.11 and B.12.

3. Basis for Relief

The automated examination of the RPV seam welds as identified in the NMP2 PSI Program Plan can only be performed on a limited scope due to Vessel Weld Transitions, RPV stabilizers, RPV ID plate, nozzles and mechanical limitations of the scanning equipment. The extent and causes of the limitations are shown on pages 3 thru 5. The integrity of the subject welds has also been previously verified by non destructive examination during fabrication under ASME Sect. III.

4. Inspection Period for Relief Request

Pre Service Inspection

5. Alternate Tests or Examinations

Ultrasonic Inspections performed in the vessel fabrication shop will also be used.

6. Schedule for Implementing Alternate Test

Previously performed in shop.

7. Impact to Overall Plant Level of Quality

Overall Plant Quality is not impacted.

8. **Preservice Examination Results**

ASME XI volumetric exam results and shop UT data will be submitted in the final report.

9. **Radiation Considerations**

None

Weld	Approximate extent and cause of limitation
AA	Portions of the 40" high exam area on either side of the weld were not covered for a total of less that 5° of the circumference due to mechanical limitations of the scanning equipment. Two areas approximately 3" wide and 13" from weld center line were not covered for less than 20° of the circumference due to interference with nozzle N-9.
AB	Portions of the 40" high exam area were not covered for a total of less than 30° of the circumference due to mechanical limitations of the scanning equipment.
AC	Portions of the 40" wide exam area were not covered for a total of less than 60° of the circumference, and an approximately 1" wide band, 4" from weld center line was not covered for less than 90° of the circumference due to mechanical limitations of the scanning equipment. Two areas approximately 10" high and 10° around circumference were also not covered due to interference with nozzle N-10.
AD	Due to congestion in the area of this upper ring girth weld there are many areas of interference with RPV stabilizers, other nozzles, vessel transition region, and mechanical limitations. The areas not covered make up approximately 50% of the total exam area.

Weld	Approximate extent and cause of limitations
BA	Portions of the 20° wide exam band were not covered for a total height of less than 40" due to interference with N2 nozzle. An area approximately 0.2° wide and 2.0° away from weld center line was not covered for a total height of less than 80" due to mechanical limitations of the scanning equipment.
BB	Portions of the 20° wide exam band were not covered for a total height of less than 45" due to nozzle interference. Areas approximately 2.0° wide, 6° away from weld center line with a height of 10" and 4.0° wide, 3° away from weld center line with a height of 30" were also not covered due to nozzle blends and mechanical limits of the scanning equipment.
BC	Portions of the 20° wide exam band were not covered for a total height of less than 35" due to nozzle interferences.
BD	An area approximately 1.0° wide and 4° from the weld center line was not covered for a total height of less than 160" due to mechanical limitations of the scanning equipment.
BE	Portions of the 20° wide exam band were not covered for a total height of less than 25" due to nozzle interferences. An area approximately 2.0° wide and 4° away from the weld center line was not covered for a total height of less than 160" due to mechanical limitations of the scanning equipment.

Weld	Approximate extent and cause of limitations
BF	Portions of the 20° wide exam band were not covered for a total height of less than 100" due to nozzle interferences and mechanical limitations of scanning equipment. An area approximately 2.0° wide and 2° from weld centerline was not covered for a total height of less than 160" also due to mechanical limitations.
BG	Portions of the 20° wide exam band were not covered for a total height of less than 35" due to RPV stabilizer interference. An area approximately 0.5° wide and 2° away from weld centerline was not covered for a total height of less than 160" due to mechanical limitations of the scanning equipment.
BH	Portions of the 20° wide exam band were not covered for a total heioght of less than 60" due to RPV stabilizer and nozzle interferences. An area approximately 0.5° wide and 2° away from weld centerline was not covered for a total height of less than 160" due to mechanical limitations of scanning equipment.
BJ	Portions of the 20° wide exam band were not covered for a total height of less than 40" due to RPV stabilizer interferences.

Nine Mile Point Unit 2

Relief Request No: RR-IWB-4

1. **Identification of Components**

   Page 3 of 3 identifies those integral attachment welds for which relief from partial ASME XI exam is required.

2. **ASME Section XI Requirements**

   Surface examinations are required for the integral attachment welds in accordance with table IWB-2500-1, Category B-K-1, Item No. B10.10.

3. **Basis for Relief**

   The examinations of the subject welds as identified in the NMP2 PSI Program Plan can only be performed on a limited scope due to permanent interferences as are indicated on Page 3 of 3. The integrity of the integral attachment welds has also been previously verified by non destructive examination during erection under ASME Sect. III.

4. **Inspection Period for Relief Request**

   Pre Service Inspection

5. **Alternate Tests or Examinations**

   ASME Sect. XI surface exams are performed. Results of ASME Sect. III surface exam performed prior to installation of permanent obstruction will also be used.

6. **Schedule for Implementing Alternate Test**

   During Pre Service Inspection

7. **Impact to Overall Plant Level of Quality**

   Overall plant quality is not impacted

8. Preservice Examination Results

ASME XI and ASME III surface exam results will be submitted in the final summary report.

9. Radiation Considerations

None

Nine Mile Point Unit 2

Relief Request No: RR-IWB-4

Due to interferences on the following welds, 100% coverage of magnetic particle examinations could not be accomplished. As a minimum of 75% of the weld exam area has been covered to full code requirements.

Weld	Interferences
2FWS-47-18-FW300 thru 301	Permanent Plate
2MSS-01-13-FW324 thru 327	Permanent Plate
2MSS-01-14-FW320 thru 323	Permanent Plate
2MSS-01-15-FW310 thru 317	Permanent Clamp
2MSS-01-15-FW320 thru 323	Permanent Plate
2MSS-01-15-FW332 thru 335	Permanent Clamp
2MSS-01-16-FW308 thru 315	Permanent Clamp

Nine Mile Point Unit 2

Relief Request No: RR-IWB-5

1. Identification of Components

Page 3 of 6 identifies those piping welds for which relief from partial ASME XI exam is required.

2. ASME Section XI Requirements

Volumetric and surface examinations are required for the piping welds in accordance with table IWB-2500-1, Category B-J, Item No. B9.11.

3. Basis for Relief

The surface examinations of the MSS and the volumetric examination of the RCS piping welds as identified in the NMP2 PSI Program Plan can only be performed on a limited scope due to permanent interferences as are indicated on Page 3 of 6. Pages 4 thru 6 of 6 show the area covered by the ASME XI Preservice examinations as well as the permanent interferences. The integrity of the piping welds has also been previously verified by non destructive examination during erection under ASME Sect. III.

4. Inspection Period for Relief Request

Pre Service Inspection

5. Alternate Tests or Examinations

ASME Sect. XI surface exams are performed. Results of ASME Sect. III volumetric exam performed prior to installation of permanent obstruction will also be used.

6. Schedule for Implementing Alternate Test

During Pre Service Inspection

7. Impact to Overall Plant Level of Quality

Overall Plant qualilty is not impacted.

8. Preservice Examination Results

ASME XI volumetric/surface and ASME III volumetric exam results will be submitted in the final summary report.

9. Radiation Considerations

None

Nine Mile Point Unit 2

Relief Request No: RR-IWB-5

Due to interferences on the following welds, 100% coverage of the surface examinations of the MSS and the volumetric examination of the RCS piping welds could not be accomplished.

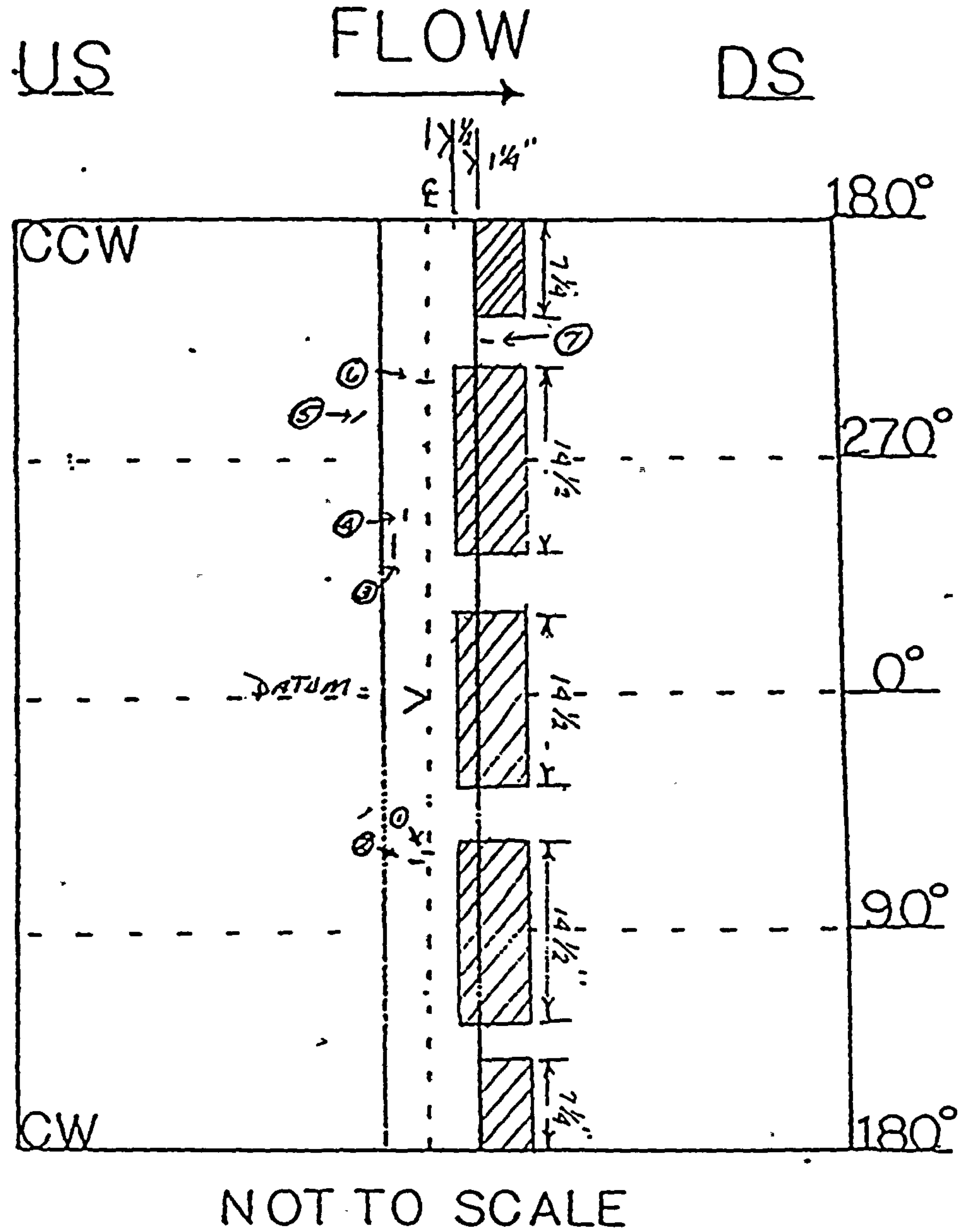
Weld	Interferences
2MSS-01-14-SW020	Permanent Integral Attachment
2MSS-01-15-SW016	Permanent Integral Attachment
2RCS-64-00-RCS-LW15	Permanent Sweepolet

SKETCH SHEET

Area/System MSS/ 01-15 Page 2 of 2

Data Sheet No. 7720-1930

Item No. 2-MSS-SW 01-15-SW016



INFORMATION ONLY

Examiner ____ Level II Date 2-1-86

Examiner N/A Level N/A Date N/A

Reviewer ____ Level III Date 2/5/86

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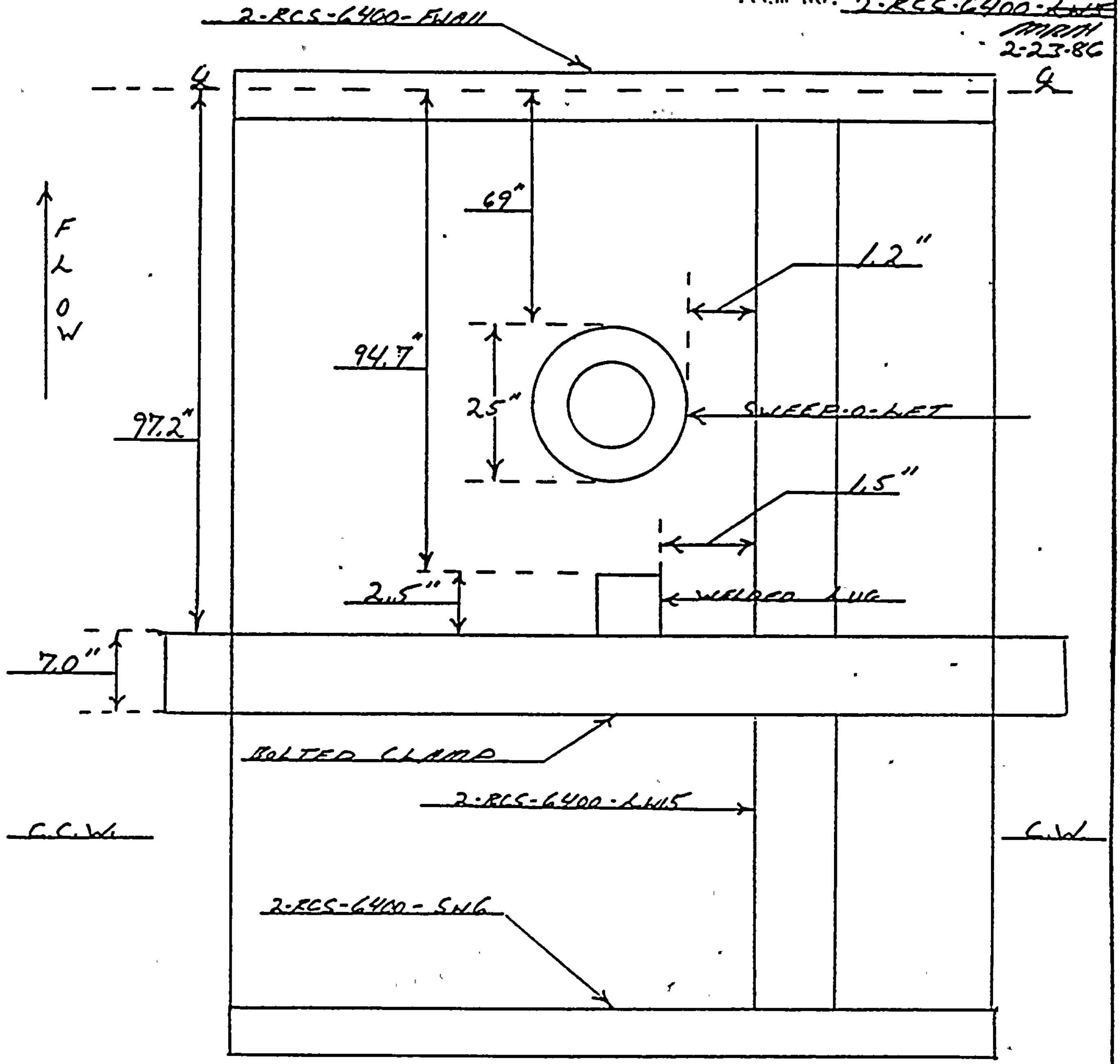
Area/System RCS / ISO# 6400 REV. 1 Page 3 of 4

NOT TO SCALE

Data Sheet No. 7718-52

Item No. 2-RCS-6400-~~LW1~~ LW15 MRM 2-23-86



NOTE: 0° SCAN LIMITED FOR A 25" AREA ON CCW SIDE DUE TO SWEEP-O-LET OBSTRUCTION.

NOTE: 0° SCAN LIMITED FOR A 2.5" AREA ON CCW SIDE DUE TO LUG OBSTRUCTION.

NOTE: 0° SCAN INCOMPLETE FOR A 7.0" AREA ON BOTH SIDES DUE TO BOLTED CLAMP OBSTRUCTION.

Examiner [signature] Level II Date 2-23-86

Examiner [signature] Level II Date ~~2-26~~ 2-23-86 MRM 2-23-86

Reviewer [signature] Level III Date 2/28/86

Nine Mile Point Unit 2

Relief Request No: RR-IWB-6

1. Identification of Components

Page 4 thru 5 identifies the stainless steel circumferential butt welds for which partial relief from ASME XI exam is required.

2. ASME Section XI Requirements

Volumetric and surface examinations are required for these piping butt welds in accordance with table IWB-2500-1, Category BJ, Item No's. B9.11, B9.31

3. Basis for Relief

The volumetric examinations of the subject welds as identified in the NMP2 PSI Program can only be performed on a limited scope due to piping system design and fitting configuration. These stainless steel welds have been examined from one side using the UT techniques specified on applicable line D or F of the attached matrix. The inspection data sheet for each specific weld defines in detail the extent of coverage obtained by the combination of angles, directions and techniques utilized. Structural integrity has also been verified during erection by volumetric and surface examinations under ASME Sect. III.

4. Inspection Period for Relief Request

Pre Service Inspection

5. Alternate Tests or Exainations

ASME XI surface exams are performed on the code required surface area of the subject welds. Latest UT techniques are employed and documented in detail in order to establish a meaningful baseline for future ISI comparisons.

6. Schedule for Implementing Alternate Test

During Pre Service Inspection

7. Impact to Overall Plant Level of Quality

Overall Plant Quality is not impacted

8. Preservice Examination Results

ASME XI surface and volumetric exam results will be submitted in the final report.

9. Radiation Considerations

None

Relief Request No.: RR-IWB-6

TABLE A

EXAMINATION MATRIX FOR STAINLESS PIPING

		Austenitic Piping 45° Refracted L	IGSCC & Inner ~~Exam~~ 45° & 60° shear 1/2 Vee	Austenitic Piping 45° Shear 2nd Leg Fusion Zone
A.	Long Seams (SW) Access Both Sides	N/A	Yes	Yes
B.	Fittings Long (SW) Access Both Sides	N/A	Yes	Yes
C.	Circ Welds (SW) Access Both Sides	N/A	Yes	Yes
D.	Circ Welds (SW) Access One Side	N/A	Yes (One Side)	Yes (One Side)
E.	Circ Welds (FW) Access Both Sides	Yes If no beam skew If Contour O.K.	Yes	Only if 45° RL is ineffective
	Circ Weld (FW) Access One Side	Yes If no beam skew If contour O.K.	Yes	Only if 45° RL is ineffective
G.	Circ Overlay Weld (SW)	N/A	Yes	Yes
H.	Circ Overlay Weld (FW)	Yes If no beam skew If contour O.K.	Yes	Only if 45° RL is ineffective

Note: The letter corresponding to the examination condition and technique shall be noted in the remarks section of the examination data sheet.

Nine Mile Point Unit 2

Relief Request No: RR-IWB-6

CAT	ITEM	WELD H	CONFIGURATION	EXAM FROM
B-J	B9.31	2RCS-64-ØØ-SW56	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	FWBØ8	Elbow to VLV	Elbow Side Only
B-J	B9.31	SW29	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	SW27	Pipe to Tee	Pipe Side Only
B-J	B9.31	SW55	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	SW58	Pipe to Cross	Pipe Side Only
B-J	B9.11	SW59	Pipe to Cross	Pipe Side Only
B-J	B9.31	SW6Ø	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.31	SW61	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	FWA16	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	FWA14	Pipe to Reduce	Pipe Side Only
B-J	B9.11	FQA19	Pipe to Safe End Ext.	Pipe Side Only
B-J	B9.31	SW28	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	SW15	Pipe to Cross	Pipe Side Only
B-J	B9.31	SW16	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.31	SW17	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	SW38	Pipe to Toe	Pipe Side Only
B-J	B9.31	SW41	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.31	SWØ1	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.31	FWA24	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	SW37	Pipe to Tee	Pipe Side Only
B-J	B9.11	FWBØ5	Elbow to Pump	Elbow Side Only
B-J	B9.11	FWB1Ø	Elbow to VLV	Elbow Side Only
B-J	B9.31	FWB24	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.31	SWØ7	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.31	SW51	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	SW12	Tee to Reducer	Reducer Side Only
B-J	B9.11	SW52	Tee to Reducer	Reducer Side Only
B-J	B9.11	FWA17	Pipe to Safe End Ext.	Pipe Side Only
B-J	B9.11	FWA18	Pipe to Safe End Ext.	Pipe Side Only
B-J	B9.11	FWA2Ø	Pipe to Safe End Ext.	Pipe Side Only
B-J	B9.11	FWA21	Pipe to Safe End Ext.	Pipe Side Only
B-J	B9.11	FWB21	Pipe to Safe End Ext.	Pipe Side Only
B-J	B9.11	FWB2Ø	Pipe to Safe End Ext.	Pipe Side Only
B-J	B9.11	FWB18	Pipe to Safe End Ext.	Pipe Side Only
B-J	B9.11	FWB17	Pipe to Safe End Ext.	Pipe Side Only
B-J	B9.11	FWBØ7	Pipe to VLV	Pipe Side Only
B-J	B9.11	FWB1Ø	Elbow to VLV	Elbow Side Only
B-J	B9.11	FWAØ5	Elbow to Pump	Elbow Side Only
B-J	B9.11	FWAØ8	Pipe to VLV	Pipe Side Only
B-J	B9.11	FWA1Ø	Elbow to VLV	Elbow Side Only
B-J	B9.11	FWAØ7	Pipe to VLV	Pipe Side Only
B-J	B9.11	FWA13	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	FWAØ3	Elbow to VLV	Elbow Side Only
B-J	B9.11	FWAØ5	Elbow to Pump	Elbow Side Only
B-J	B9.11	FWBØ3	Elbow to VLV	Elbow Side Only
B-J	B9.11	FWBØ9	Pipe to VLV	Pipe Side Only

Nine Mile Point Unit 2

Relief Request No: RR-IWB-6

CAT	ITEM	WELD H	CONFIGURATION	EXAM FROM
B-J	B9.11	2RCS-64-00-FWA09	Pipe to VLV	Pipe Side Only
B-J	B9.11	FWB04	Pipe to VLV	Pipe Side Only
B-J	B9.11	FWB20	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	FWA12	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	FWB12	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	FWB13	Pipe to Sweep-o-let	Pipe Side Only
B-J	B9.11	FWB19	Pipe to Safe End Ext.	Pipe Side Only
B-J	B9.11	FWA01	Pipe to Safe End Ext.	Pipe Side Only
B-J	B9.11	FWB01	Pipe to Safe End Ext.	Pipe Side Only
B-F	B5.50	2WCS-09-05-SW028	Pipe to Tee	Tee Side Only
B-J	B9.11	SW030	Pipe to Tee	Tee Side Only
B-J	B9.11	SW032	Pipe to Tee	Tee Side Only
B-J	B9.11	SW033	Pipe to Flange	Pipe Side Only
B-J	B9.11	SW025	Pipe to Flange	Pipe Side Only

NINE MILE POINT UNIT 2

RELIEF REQUEST NO: RR-IWB-7

1. Identification of Components

Reactor Vessel Top & Bottom head welds # RPV-DA,DB,DC,DD,DE, DF,DG,DR and, RPV-AG require partial relief from ASME XI examination.

2. ASME Section XI Requirements

Volumetric examination is required for the bottom head welds in accordance with table IWB 2500-1, category BA, item no. B1.21, B1.22. Volumetric and surface examination is required for top head to flange per stem no. B1.40.

3. Basis for relief

Manual examination of these RPV bottom head welds as identified in the NMP2 PSI program plan can only be performed on a limited scope due to CRD penetrations and vessel support skirt. Only Approx. 12" to 24" on each end of welds RPV DG and DR can be examined due to interface with CRD penetrations housings. Approximately one foot is not being examined on each of welds RPV DA thru DF due to interference with RPV support skirt. Weld # RPV-AG is top head to flange weld which can only be examined from the head side due to flange configuration. Structural integrity of these welds has also been verified during fabrication under ASME Sect. III.

4. Inspection Period for Relief Request

Pre Service Inspection

5. Alternate Tests or Examinations

Additional angles are used for volumetric exams and weld RPV-AG receives a surface exam over the entire code required surface area.

6. Schedule for Implementing Alternate Test

During Pre Service Inspection

7. Impact to Overall Plant Level of Quality

Overall plant Quality is not affected.

8. Preservice Examination Results

ASME XI volumetric and surface exam results will be submitted in the final summary report.

9. Radiation Considerations

None

NINE MILE POINT UNIT 2

RELIEF REQUEST NO.: RR-IWB-8

1. Identification of Components

   Page 2 identifies recirculation System welds for which partial relief from ASME XI exam is required.

2. ASME Section XI Requirements

   Volumetric and Surface examinations are required for these recirculation system welds in accordance with table IWB-2500-1, category BJ, item no. B9.11.

3. Basis for Relief

   The volumetric examinations of the subject welds as identified in the NMP2 PSI program plan can only be performed on a limited scope due to varying degrees of austenitic weld overlays. The ultrasonic responses encountered while performing examinations are described in the attached report. The inspection data sheet for each specific weld defines in detail the extent of coverage obtained for use as a baseline for future ISI comparison. Other welds in the system that are subject to the same operating conditions receive complete ASME XI volumetric examinations. Structural integrity has also been verified during erection by volumetric and surface examination under ASME sect III.

4. Inspection Period for Relief Request

   Pre Service Inspection

5. Alternate Tests or Examinations

   ASME XI surface exams are performed on the code required surface area of the subject welds. Latest UT techniques, described in the attached report, are employed and documented in detail.

6. Schedule for Implementing Alternate Test

   During Pre Service Inspections

7. Impact to Overall Plant Level of Quality

   Overall Plant Quality is not impacted

8. Preservice Examination Results

   ASME XI surface and volumetric exam results will be submitted in the final report.

9. Radiation Considerations

   None

NINE MILE POINT UNIT 2

RELIEF REQUEST NO: RR-IWB-8

WELD NUMBERS

2-RCS-64-ØØ-FWB 11
2-RCS-64-ØØ-FWA Ø6
2-RCS-64-ØØ-FWB Ø6
2-RCS-64-ØØ-FWA 21
2-RCS-64-ØØ-FWA 2Ø
2-RCS-64-ØØ-FWA 19
2-RCS-64-ØØ-FWA 18
2-RCS-64-ØØ-FWA 17
2-RCS-64-ØØ-FWB 17
2-RCS-64-ØØ-FWB 18
2-RCS-64-ØØ-FWB 19
2-RCS-64-ØØ-FWB 2Ø
2-RCS-64-ØØ-FWB 21

ULTRASONIC EXAMINATION OF RECIRCULATION LINE STAINLESS STEEL OVERLAYED WELDS NINE MILE POINT, UNIT 2

This report serves to document the information from ultrasonic findings with regard to the Metallurgical nature of the weldments contained within the recirculation loops. (see list attached)

Welds within the recirculation loops have been overlayed with welding (an example is shown in Figure 1). During the examination of these welds, it was noted that in certain welds the intended angle at which the sound should travel was not the actual angle observed.

It appears that the ultrasound has a tendency to divert from it's intended path and redirect in an almost perpendicular fashion to the inside wall (ID) of the pipe. The generally accepted theory pertaining to this type of occurence is that the columnar grain structure present in Austenitic weldments provides a "wave guide" effect and thus carries the sound in a direction other than the intended one. Other theories such as granular impedence or filtration have also been postulated.

Examinations

These were performed in accordance with the appropriate procedure. Beam redirection was noted and where this occurence was evident, other examination frequencies and angles were used to try and overcome the effects produced by the grain structure. A low frequency was selected because of it's longer wavelength and greater penetration abilities.

1. 45° x 1.5 MHz shear wave search units were initially applied with little success. Reflections which when plotting at the calibration measured 45° angle, appeared to occur at approximately 3/4 'T' metal path. These reflections when postulated perpendicular to the surface, occur at or around 'T' 0° (Velocity Shear Wave). Counterbore could be detected but this appeared almost directly beneath the search unit, confirming beam redirection.

2. 60° x 1.5 MHz shear wave was selected and applied as above. The results noted with this unit were not unlike those noted when using the 45° shear wave unit. Again the reflection observed, appeared to originate from the ID surface.

3. The frequency was then reduced to 1.0 MHz with much the same result as in 1 + 2 above.

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4. Refracted longitudinal techniques were applied. The rationale behind this excerise was consideration to the fact that (a) A longer wavelength can be achieved for a given frequency. (b) Penetration should be greater due to (a) above and (c) It has been demonstrated in the past in similar situations, longitudinal modes are less prone to beam redirection than are shear modes.

   The disadvantage is that while using longitudinal wave modes, the response to corner reflectors (cracks) is less desirable than the response noted when using shear wave modes. (This is due to mode conversion and energy losses in a corner situation using longitudinal wave modes.) We essentially have a "trade-off" situation.

   4.1 One of the other problems generated is that because of the incident angle necessary to produce a refracted longitudinal wave mode in the material, the "noise" generated in the search unit (SU) is greater than that in a shear wave search unit. To overcome this, a transmit/receive unit is used. Here again there is a trade off in that, for a given size SU the element size has to be smaller resulting in a greater beam divergence for a given frequency. This reduces the amount of energy that is transmitted into the material.

   4.2 With these and other considerations in mind, this technique was applied at code calibration sensitivity which resulted in excessive amounts of noise returning to the SU from within the material. To add to this, it was discovered that the beam redirection noted when using the shear wave techniques, also occurred when using the refracted "L" Wave techniques.

   4.3 An interesting observation was that the redirection is not necessarily the same when facing the sound "beam" in opposite directions. For example when facing the SU (on a vertical pipe) in the upward direction, beam redirection was noted to be considerably greater than when rotating the search unit through 180° and facing the beam downward. We can readily assume that this has to do with the direction of the columnar grains (which follow the direction of heat dissippation during their formation and generally grow epitaxially from weld bead to weld bead). We can also assume that in a vertically welded situation, the structure will differ considerably from that welded in a horizontal situation, basically determining that the responses observed should be weld direction sensitive.

5. To unquestionably verify the above, a variable angle search unit was applied. This unit is a 2 1/4 MHz transducer mounted on a device which enables the sound to be introduced into the material at any

selected angle. 0° longitudinal wave was the starting point, with the unit mounted on the overlayed area on FWB11 facing upwards, (toward the weld) the instrument calibrated in metal path for longitudinal velocity.

5.1 A back reflection (BR) and repeat BR's were apparent on the CRT. The unit was scanned forward and the BR appeared constant until the counterbore (CB) was located, at which time the metal path changed accordingly. The unit was replaced to its original position and the BR returned to its original position on the time base. The unit was then angled to produce a refracted longitudinal wave. The BR signal amplitude was seen to reduce and a second reflection (CB) could be seen appearing just after the BR (later in time). The unit was angled over until the second signal was at peak amplitude at which time the BR could no longer be seen. The unit was scanned forward toward the CB and the signal moved closer in time until it disappeared. This sequence was repeated, each time with a steeper angle. It is noteworthy that regardless of the angle introduced into the material, the CB always appeared at or slightly after 'T" 0°. This confirmed the fact that the beam was not being reflected in the manner in which it should be, given "normal" conditions.

5.2 At some point as the angle was increased, a series of signals could be seen later in time as the unit was scanned back and forth. These signals appeared at or about 'T" 0° for shear wave and increased in amplitude as the beam angle was increased. This was established to be shear wave redirection (due to its position in time on the time base).

5.3 The unit could indicate that higher angles, beam redirection may be more evident for a given grain structure. Similar results were observed while going through the "longitudinal wave" range.

5.4 The unit was rotated through 180° and the above was repeated. This time beam redirection was minimal as noted above in 4.3. The absense of 'T' 0° signals would indicate that there is not significant redirection while scanning in this direction (facing away from the weld (down)). A·Pitch/Catch using 45° shear wave also performed in this (downward) direction and a "full vee path" could be detected at a measured and calculated angle of approximately 43°, which would tend to substantiate the conclusion that the sound is extremely sensitive to the dendritic formation angle, and in this case is redirecting mainly when scanning with the beam directed upwards toward the weld.

6. The possibility of introducing large amounts of low frequency energy was considered, and a dual 1 MHz x 1" diameter longitudinal wave SU was applied (each side having a 1" diameter element). The unit had

a "roof angle" of approximately 2° and a forward refracted angle of 45° in the material under test. Beam redirection was still apparent, but due to the large energy source, return signals were noted. These were calculated as occurring from ID geometry at an angle of approximately 45°. the "prose" were that we were now penetrating the material at a known angle. The cons were more in evidence. The search unit being so large and the surface undulations being such as they are, contact was made and lost too frequently to perform a meaningful examination. With this condition, the beam shape characteristics change due to variations in contact. Considering the small area available (due to physical geometric constraints - the coverage and information acquired would be marginal in terms of calling the examination "meaningful" with this unit).

Recommendations

In cases where 45° longitudinal examinations have not been attempted, these should be carried out where possible. The results should be documented and included with the existing data. We determine that based on the above exercise, the returns for effort in terms of ALARA and ultimate defect detectability will be marginal in some cases and request for relief from examination of specific welds be sought.

NES is constantly researching new techniques and technology and as developments occur, these will be made known to the utility.

Michael L. Shakinovsky

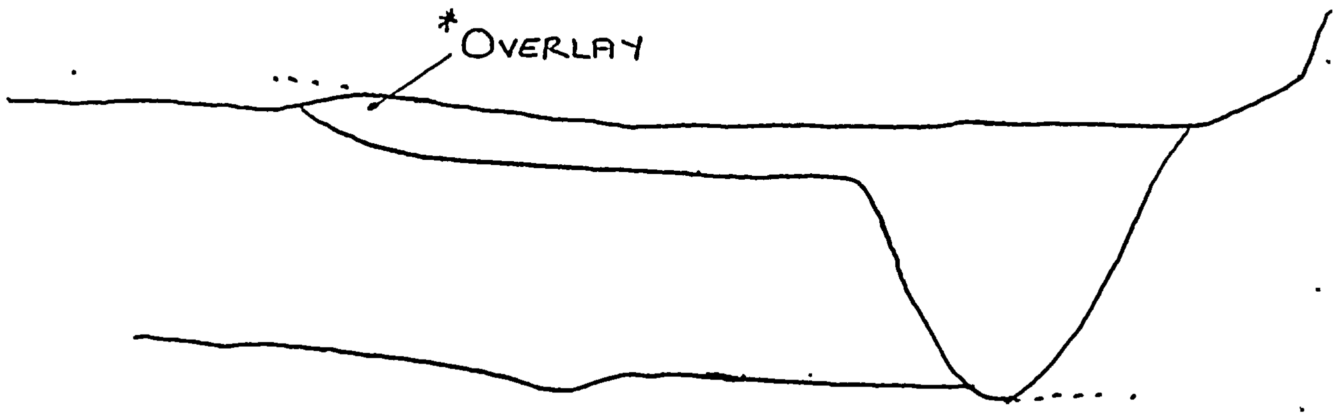
LTB

SKETCH SHEET

Area/System ______________________________ Page ___ of ___

Data Sheet No. ______________

Item No. ______________



*ESTIMATED THICKNESS BASED ON ACTUAL MEASUREMENT FROM EDGE

WELD PROFILE - ULTRASONIC + PIN GAGE

FIGURE 1

Nine Mile Point Unit 2

Relief Request No: RR-IWB-9

1. Identification of Components

Page 3 identifies RPV nozzle to safe end welds for which partial relief from ASME XI exam is required.

2. ASME Section XI Requirements

Volumetric and surface examinations are required for RPV nozzle to safe end welds in accordance with table IWB-2500-1, Category BF, Item No. B5.10.

3. Basis for Relief

The automated examination of the subject welds as identified in the NMP2 PSI Program Plan can only be performed on a limited scope. The approximate extent of coverage due to physical limitations such as nozzle blend, insulation supports and bio shield wall is shown on pages 3 and 4. Other limitations due to the inability of the examination to distinguish the weld root from the inside diameter notch on the calibration standard are detailed in the attached discussion. Structural integrity has also been verified by volumetric and surface during erection under ASME Sect. III.

4. Inspection Period for Relief Request

Pre Service Inspection

5. Alternate Tests or Examinations

ASME XI surface exams are performed on the code required surface area of the subject welds. The UT techniques described in the attached discussion are employed and documented in detail. Pictorial data is permanently stored on UDRP5 as a baseline for direct ISI comparison.

6. Schedule for Implementing Alternate Test

During Pre Service Inspections

7. Impact to Overall Plant Level Quality

Overall Plant Quality is not impacted

8. Preservice Examination Results

ASME XI surface and volumetric examination results will be submitted in the final report.

9. Radiation Considerations

None

Nine Mile Point Unit 2

Relief Request No: RR-IWB-9

WELD #	NOZZLE #	EXTENT OF COVERAGE PERP.	PARALLEL	CAUSE OF LIMITATION
RPV-KB01	N1A	64.85	100	Nozzle Blend
RPV-KB02	N1B	75.35	85.21	Nozzle Blend
RPV-KB03	N2A	71.25	94.30	Nozzle Blend
RPV-KB04	N2B	76.6	100	Nozzle Blend
RPV-KB05	N2C	74.3	91.4	Nozzle Blend
RPV-KB06	N2D	76.6	100	Nozzle Blend
RPV-KB07	N2E	69.0	100	Nozzle Blend
RPV-KB08	N2F	76.6	100	Nozzle Blend
RPV-KB09	N2G	76.6	100	Nozzle Blend
RPV-KB10	N2H	64.4	67.9	Nozzle Blend
RPV-KB11	N2J	76.6	100	Nozzle Blend
RPV-KB12	N2K	68.2	76.8	Nozzle Blend
RPV-KB17	N4A	66.1	75.5	Nozzle Blend,Insulation Suppt.
RPV-KB18	N4B	61.7	74.8	Nozzle Blend,Insulation Suppt.
RPV-KB19	N4C	22.7	22.5	Nozzle Blend,Insulation Suppt.
RPV-KB20	N4D	75.9	94.4	Nozzle Blend,Insulation Suppt.
RPV-KB21	N4E	45.3	40.1	Nozzle Blend,Insulation Suppt.
RPV-KB22	N4F	56.0	66.5	Nozzle Blend,Insulation Suppt.
RPV-KB23	N5A	36.5	43.7	Nozzle Blend,Insulation Suppt.
RPV-KB24	N6A	40.2	58.9	Nozzle Blend,Insulation Suppt.
RPV-KB25	N6B	45.2	54.3	Nozzle Blend
RPV-KB26	N6C	60.5	61.8	Nozzle Blend
RPV-KB32	N16A	30.9	31.1	Nozzle Blend,Bio Shield Wall

Nine Mile Point Unit 2

Relief Request No: RR-IWB-9

DISCUSSION

Relief on the limited examination volume of the 45° L axial examination is required:

- The limited volume is the perpendicular examination of the inner 1/3T (Approx.), to 1/2" on both sides of the weld centerline. This volume was scanned and recorded; however, the ability to evaluate is minimal due to signals from the weld root. The pictorial data from this area is preserved on UDRPS as a baseline for direct comparison to ISI data.

During the 45° perpendicular exam on both calibration blocks (N1 & N2) the indication from the notch could not be distinguished from the root indication. Both calibration blocks had the weld root ground off for just a long enough distance to put in the notch. Even though we are using a 45° longitudinal wave there are also some dissimilar material and beam skew indications. The beam skew indications are the result of dendrites, and can occur at a depth of from (Ø.7)(T) to beyond (T), whenever the ultrasonic beam enters the weld in the 45° longitudinal axial examination.

Since this examination canot distinguish weld root from the ID (inside diameter) notch, we can not "size" to code requirements in the root area of the weld.

The Ø° examination establishes the existence and location of ID Geometry; however, dissimilar materials, because of their different velocities, sometimes show up as slight thickness changes. It is therefore possible to establish if a particular angle beam indication is <u>probably</u> coming from Geometry or a dissimilar metal interface.

Manual examination can not reduce any detection or discrimination problems. Special manual techniques may help in sizing specific indicators.

The additional "unlimited" examinations performed on this volume are:

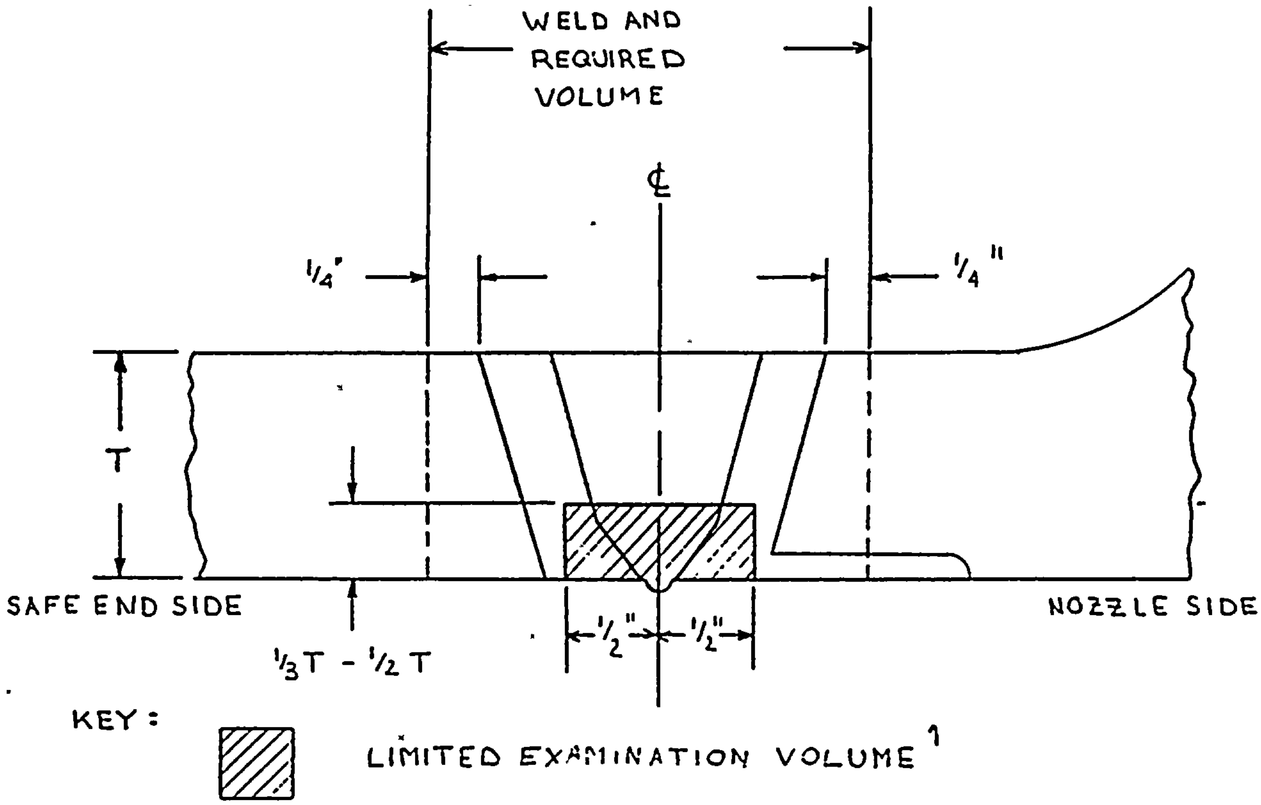
- The 45°L parallel examination with the sensitivity increased to provide a noise level suitable for IGSCC baseline data.
- A perpendicular baseline IGSCC examination covering the inner 1/3 T in the safe end material with a 52° shear wave.
- The specific weld inspection data sheet defines in detail the extent of coverage obtained from each exam performed.

SKETCH SHEET

Area/System SAFE END WELDS Page ___ of ___

EXTENT OF AXIAL COVERAGE Data Sheet No. N/A

FULL VOLUME EXAMINATION Item No. N/A



1 THIS "LIMITED VOLUME" WAS SCANNED AND RECORDED; HOWEVER, THE ABILITY TO EVALUATE THIS AREA IS MINIMAL DUE TO SIGNALS FROM THE WELD ROOT.

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Nine Mile Point Unit 2

Relief Request No: RR-IWC-7

1. Identification of Components

Page 3 of 4 identifies those RDS integral attachment welds for which relief from partial ASME XI exam is required.

2. ASME Section XI Requirements

Surface examinations are required for the RDS integral attachment welds in accordance with table IWC-2500-1, Category C-C, Item No. C3.40.

3. Basis for Relief

The examinations of the subject welds as identified in the NMP2 PSI Program Plan can only be performed on a limited scope due to the interferences from the tube steel of their associated supports. The sketch on Page 4 of 4 shows the typical configuration of the RDS integral attachment welds as well as the area covered by the ASME XI preservice surface exams and the limiting permanent interferences. The as installed position of the tube steel relative to the lug is required for the support to perform its design function. It is not practical to redesign the interfering tube steel outside the lug weld exam area or to cut the tube steel to perform this exam. The integrity of the RDS integral attachment welds has also been previously verified by non destructive examination during erection under ASME Sect. III.

4. Inspection Period for Relief Request

Pre Service Inspection

5. Alternate Tests or Examinations

ASME Sect. XI surface exams are performed. Results of ASME Sect. III magnetic particle exam performed prior to installation of permanent obstruction will also be used.

6. Schedule for Implementing Alternate Test

During Pre Service Inspection

7. Impact to Overall Plant Level of Quality

Overall Plant Quality is not impacted

8. **Preservice Examination Results**

   ASME XI and ASME III surface exam results will be submitted in the final summary report.

9. **Radiation Considerations**

   None

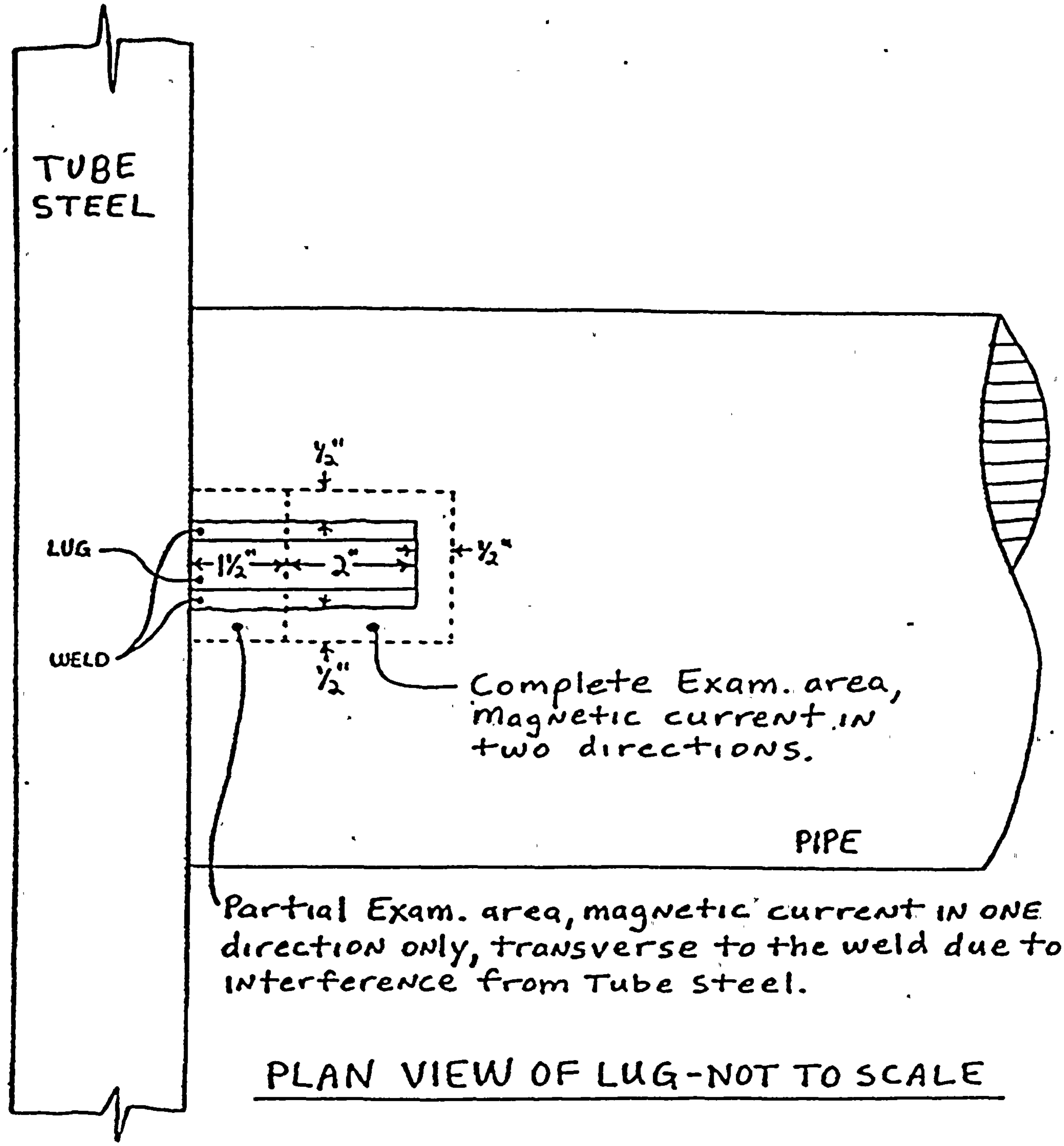
Nine Mile Point Unit 2

Relief Request No: RR-IWC-7

Due to Component support on the following welds, 100% coverage of Magnetic Particle Examinations could not be accomplished.

2-RDS-65-00-IAW01B-12 thru 19
IAW06A-12 thru 19
IAW03-12 thru 19
IAW07B-12 thru 19
IAWSP-2N-1 thru 8
IAWSP-2S-1 thru 8
IAW04B-12 thru 19
IAW22A-16 thru 25
IAW12A-20 thru 27
IAW12A-30 and 31
IAW13B-20 thru 27
IAW13B-32 and 33
IAW09A-16 thru 25
IAW15A-20 thru 27
IAW15A-30 and 31
IAW16B-20 thru 27
IAW16B-32 and 33
IAW19B-20 thru 25
IAW19B-32 and 33
IAW10B-16 thru 23

Typical for all lugs listed



PLAN VIEW OF LUG-NOT TO SCALE

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Nine Mile Point Unit 2

Relief Request No: RR-IWC-8

1. Identification of Components

Page 3 of 13 identifies those integral attachment, piping and valve body welds for which relief from partial ASME XI exam is required.

2. ASME Section XI Requirements

Surface examinations are required for the integral attachment and valve body welds ina ccordance with table IWC-2500-1, Category C-C, Item No. C3.40 and Category C-G, Item No. C6.20 respectively. Surface and volumetric examinations are required for weld 2RHS-66-57-SW005 in accordance with table IWC-2500-1, Category C-F, Item No. C5.11

3. Basis for Relief

The surface examinations of the subject welds as identified in the NMP2 PSI Program Plan can only be performed on a limited scope due to permanent interferences as are indicated on Page 3 of 13. Page 4 thru 13 of 13 shows the area covered by the ASME XI Preservice examinations as well as the permanent interferences. The integrity of the subject welds has also been previously verified by nondestructive examination during erection under ASME Sect. III.

4. Inspection Period for Relief Request

Pre Service Inspection

5. Alternate Tests or Examinations

ASME Sect.XI surface exams are performed. Results of ASME Sect. III volumetric exam for 2RHS-66-57-W005 and ASME Sect. III surface exam for the integral attachment and valve body welds performed prior to installation of permanent obstruction will also be used.

6. Schedule for Implementing Alternate Test

During Pre Service Inspection

7. Impact to Overall Plant Level of Quality

Overall Plant Quality is not impacted.

8. Preservice Examination Results

ASME XI and ASME III volumetric and or surface will be submitted in the final summary report.

9. Radiation Considerations

None

Nine Mile Point Unit 2

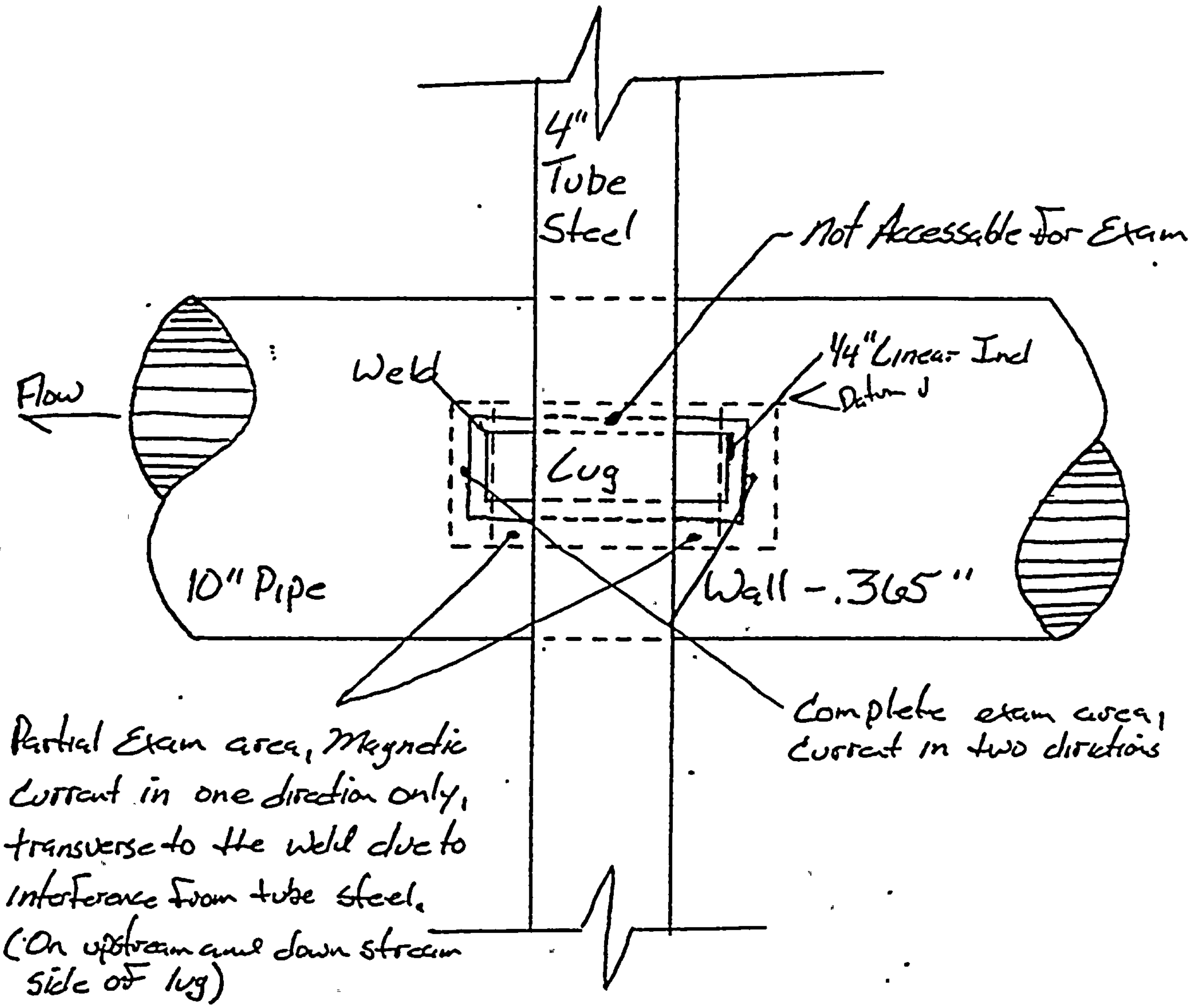
Relief Request No: RR-IWC-8

Due to interferances on the following welds, 100% coverage of surface examinations could not be accomplished.

Weld	Exam Category	Interferances
2RHS-66-57-FW307	C-C	Permanent Tube Steel
2RHS-66-57-SW005	C-F	Configuration of Flange
VWMOV1C-B thru D	C-G	Permanent Stiffener Plate
VWMOV2A-A thru C	C-G	Permanent Stiffener Plate
VWMOV112-B thru D	C-G	Permanent Stiffener Plate

SKETCH SHEET

Item No 2-RHS-6657-FW307



INFORMATION ONLY

Examiner Cal [signature] Level II Date 2-11-86

Examiner N/A Level N/A Date N/A

Reviewer [signature] Level III Date 2/20/86

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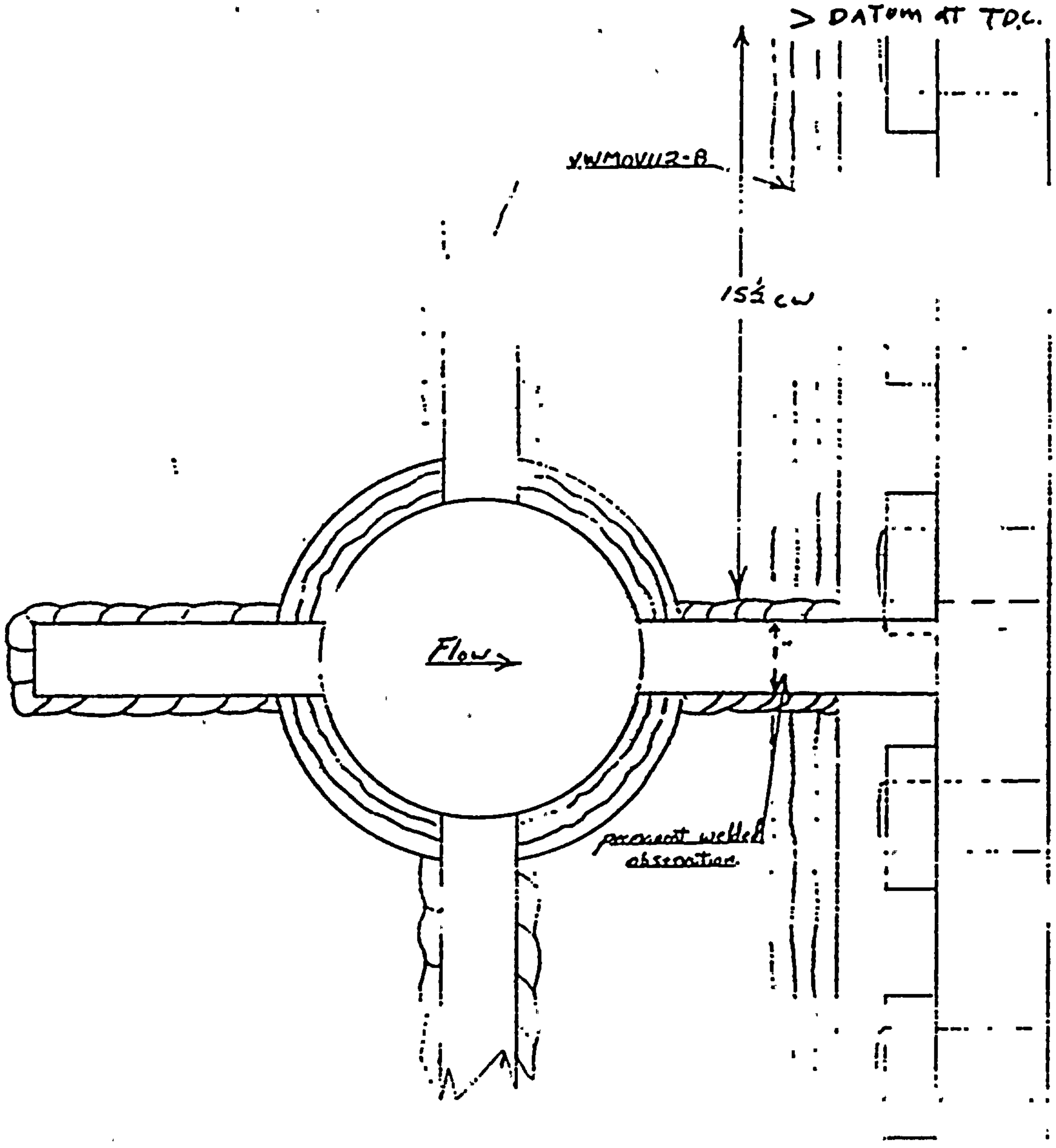
SKETCH SHEET

Area/System CSL

Data Sheet No. 7721-134

Item No. 2-CSL-26-01-VWMOV112-B



INFORMATION ONLY

Examiner ______ Level II Date 1-11-86

Examiner N/A Level N/A Date N/A

Reviewer ______ Level III Date 1/13/86

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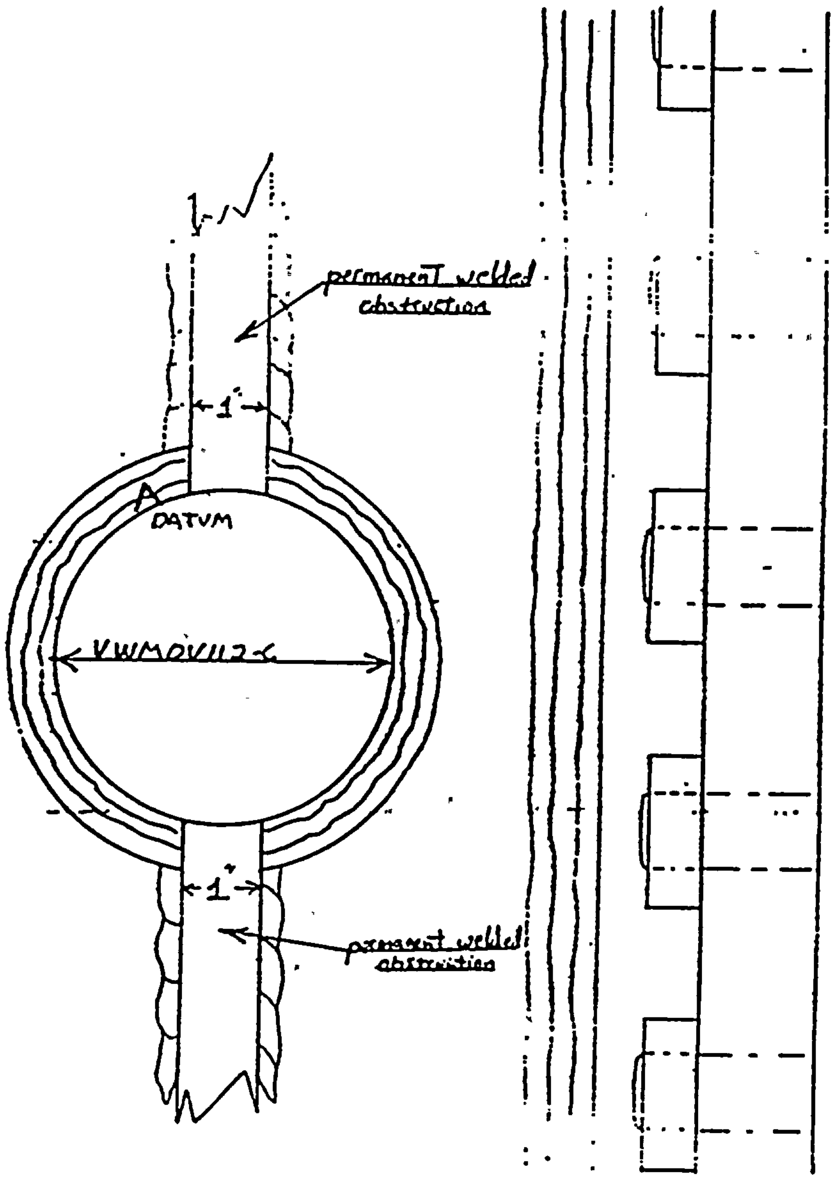
SKETCH SHEET

Area/System CSL / ISI 2607 / ISI Sketch #015 Page 2 of 9

Data Sheet No. 7721-135

Item No 2-CSL-2607-VWMOV112-C



INFORMATION ONLY

Examiner Andrew W Moss Level II Date 1-11-86

Examiner N/A Level N/A Date N/A

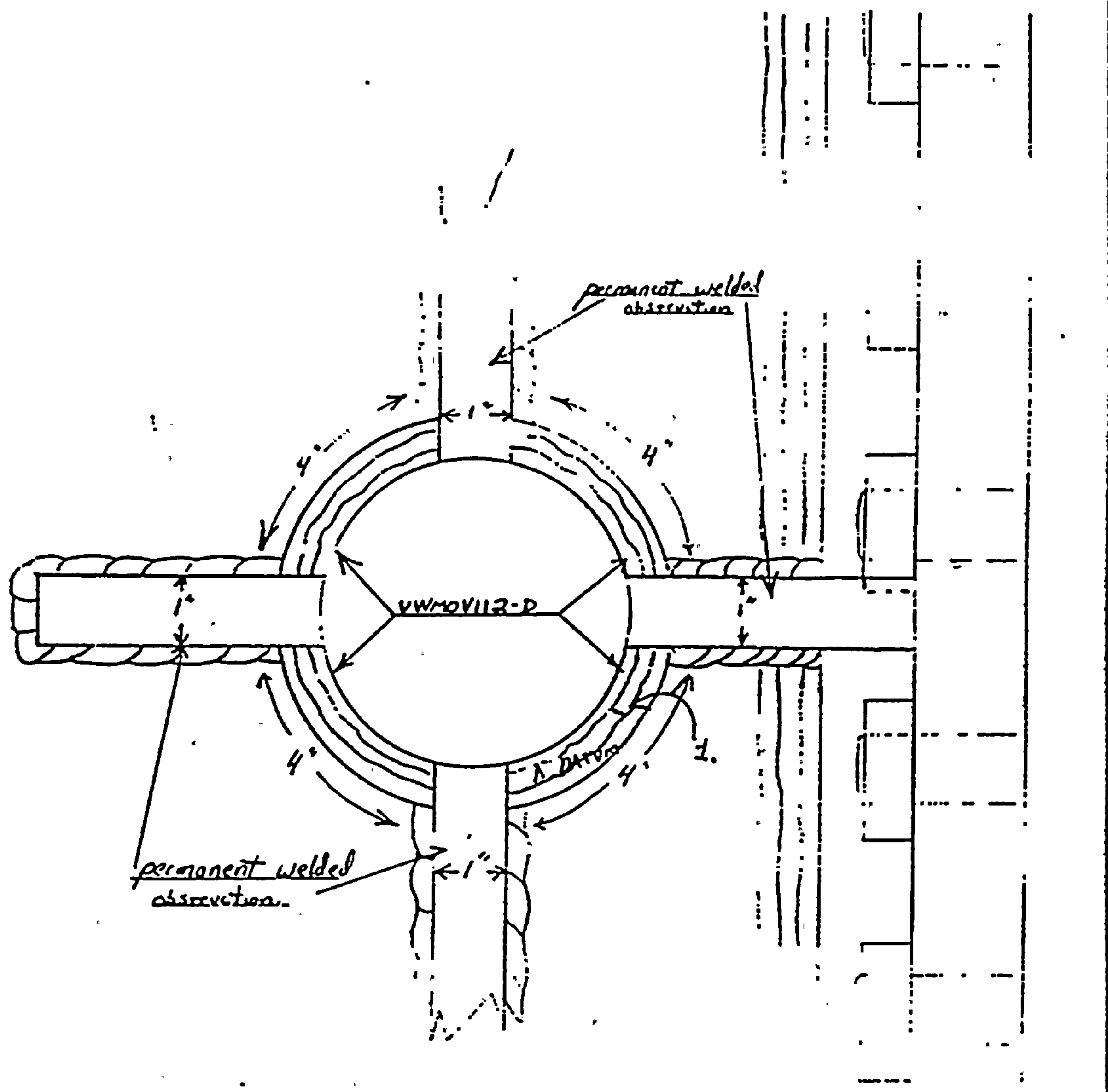
Reviewer Vit D Malon Level III Date 1/13/86

SKETCH SHEET

Area/System CSI / ISI-26-01 Rev 1 / ISI Sketch #015 Page 2 of 2

Data Sheet No. 2221-136

Item No 2-CSI-26-01-VWMOV112-D



INFORMATION ONLY

Examiner Andy W Moran Level II Date 1-11-86

Examiner N/A Level N/A Date N/A

Reviewer [signature] Level III Date 1/13/86

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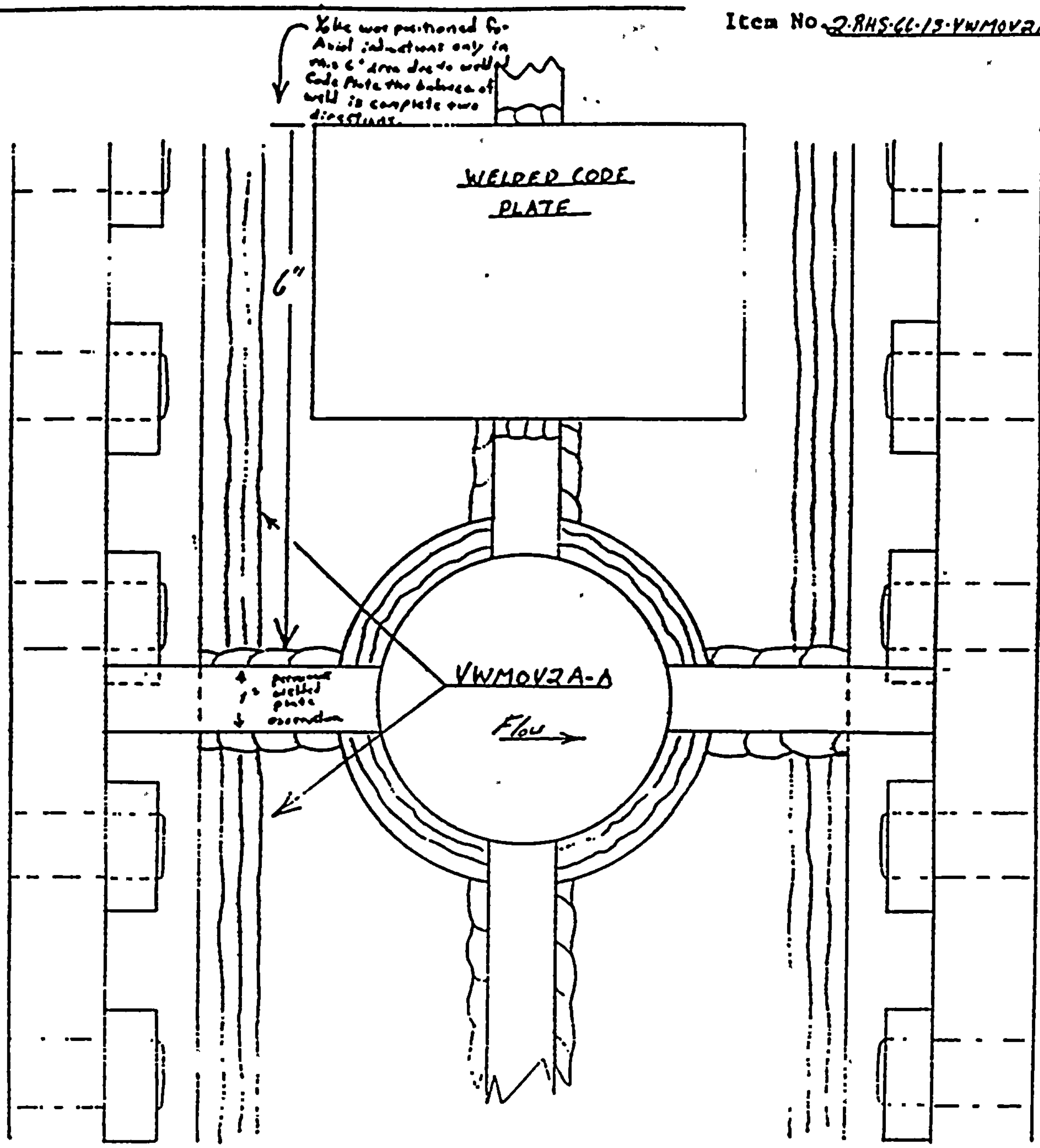
SKETCH SHEET

Area/System *RHS / ISI 66-13 Rev 1 / ISI Sketch # 104* Page *2* of *2*

Data Sheet No. *7721-113*

Item No. *2-RHS-66-13-VWMOV2A-A*



INFORMATION ONLY

Examiner *[signature]* Level *II* Date *1-9-86*

Examiner *N/A* Level *N/A* Date *N/A*

Reviewer *[signature]* Level *III* Date *2/12/86*

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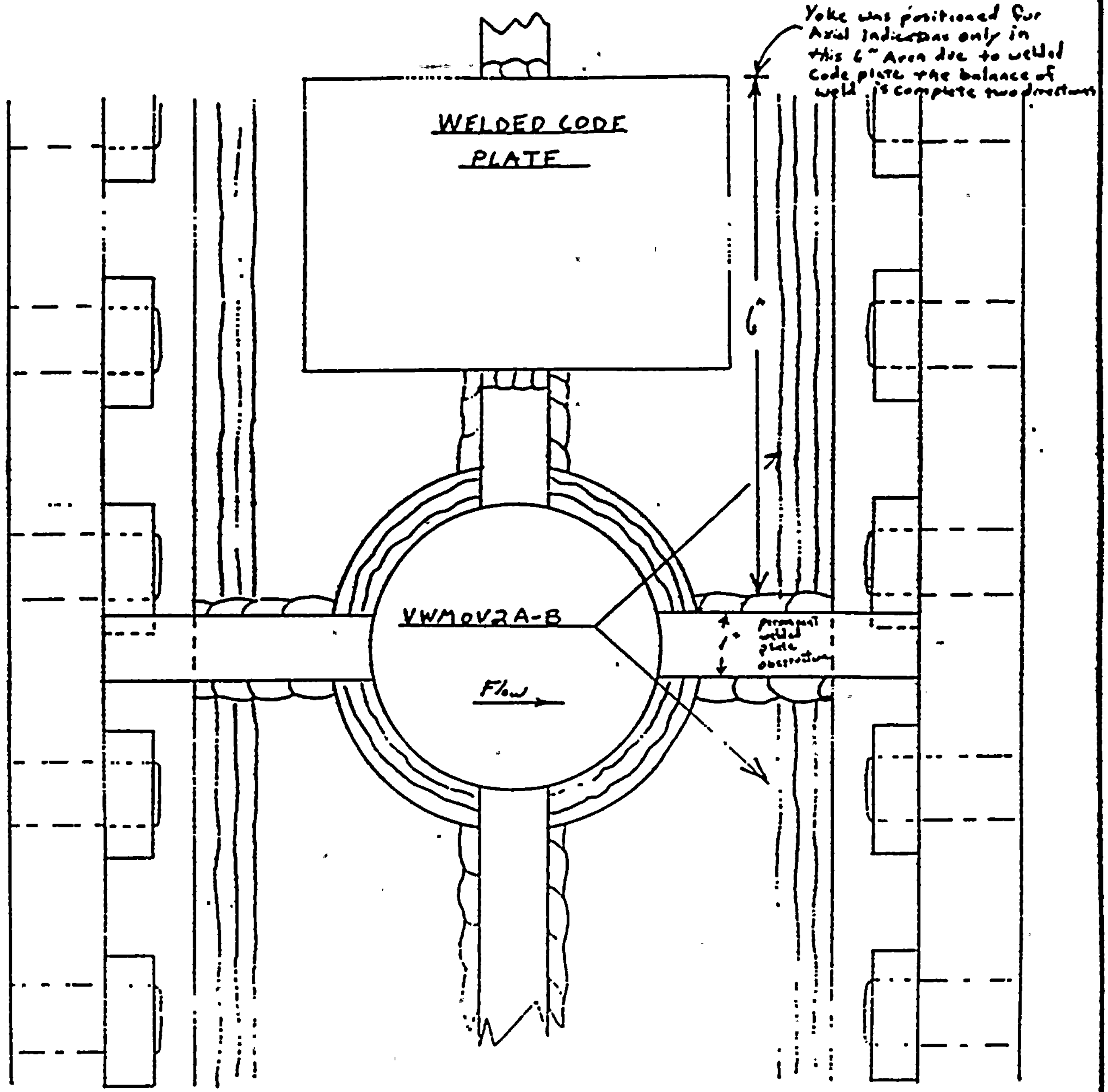
SKETCH SHEET

Area/System RHS / IST 66-13 Rev 1 / IST Sketch # 104

Page 2 of 2

Data Sheet No. 7721-114

Item No. 2-RHS-66-13-VWMOV2A-B



INFORMATION ONLY

Examiner Andre U Moore Level II Date 1-9-86

Examiner N/A Level N/A Date N/A

Reviewer [illegible] Level Date 1/12/86

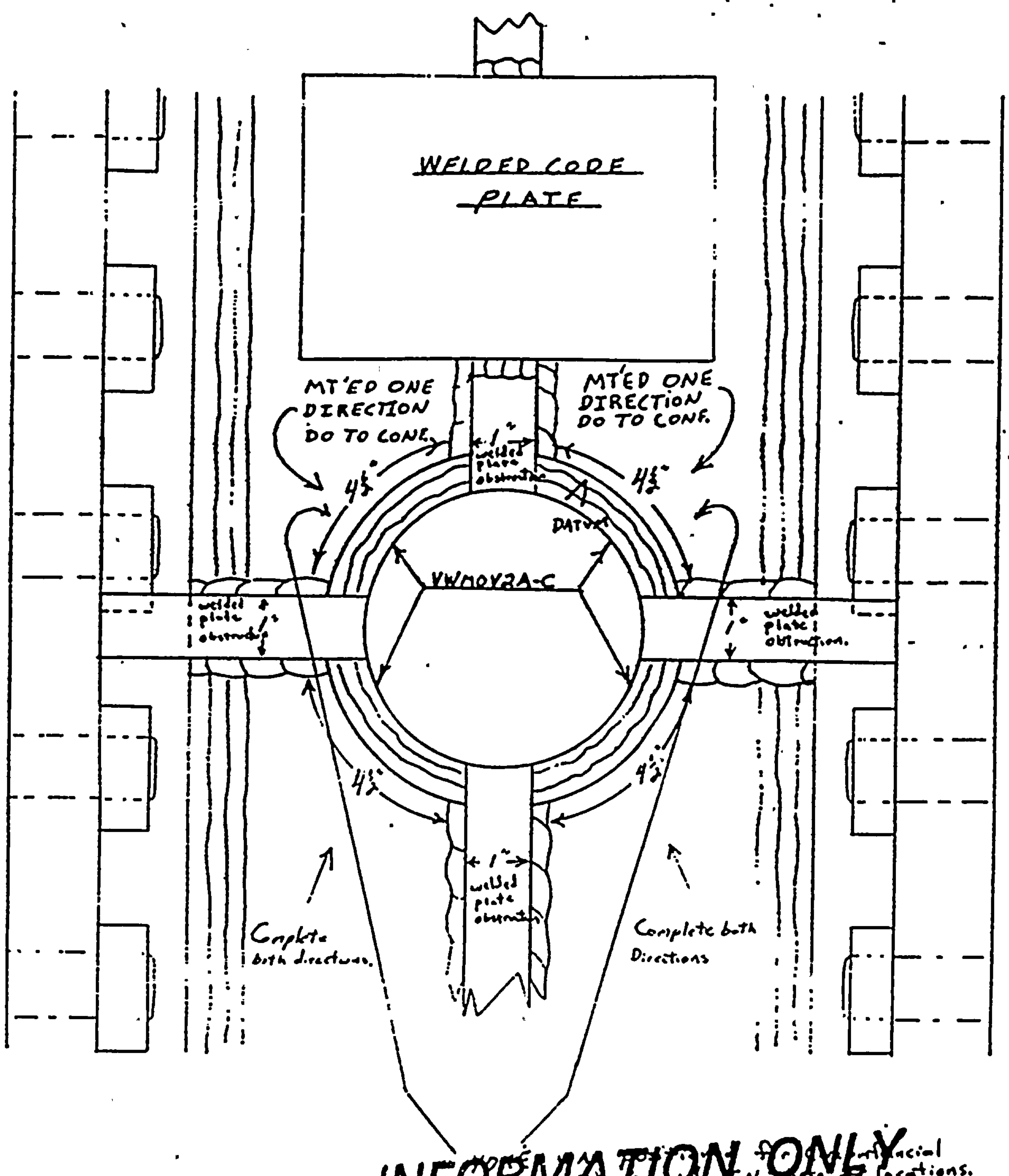
NUCLEAR ENERGY SERVICES, INC.

Nine Mile Point Unit 2 **SKETCH SHEET** Relief Request No: RR-IWC-8

Area/System RHS / ISI 66-13 Rev 2 / ISI Sketch # 104 **Page** 2 **of** 2

Data Sheet No. 7721-115

Item No. 2-RHS-66-13-VWMOV2A-C



INFORMATION ONLY

Examiner *Andr W Moore* Level II Date 1-9-86

Examiner N/A Level N/A Date N/A

Reviewer *[signature]* Level III Date 1/10/86

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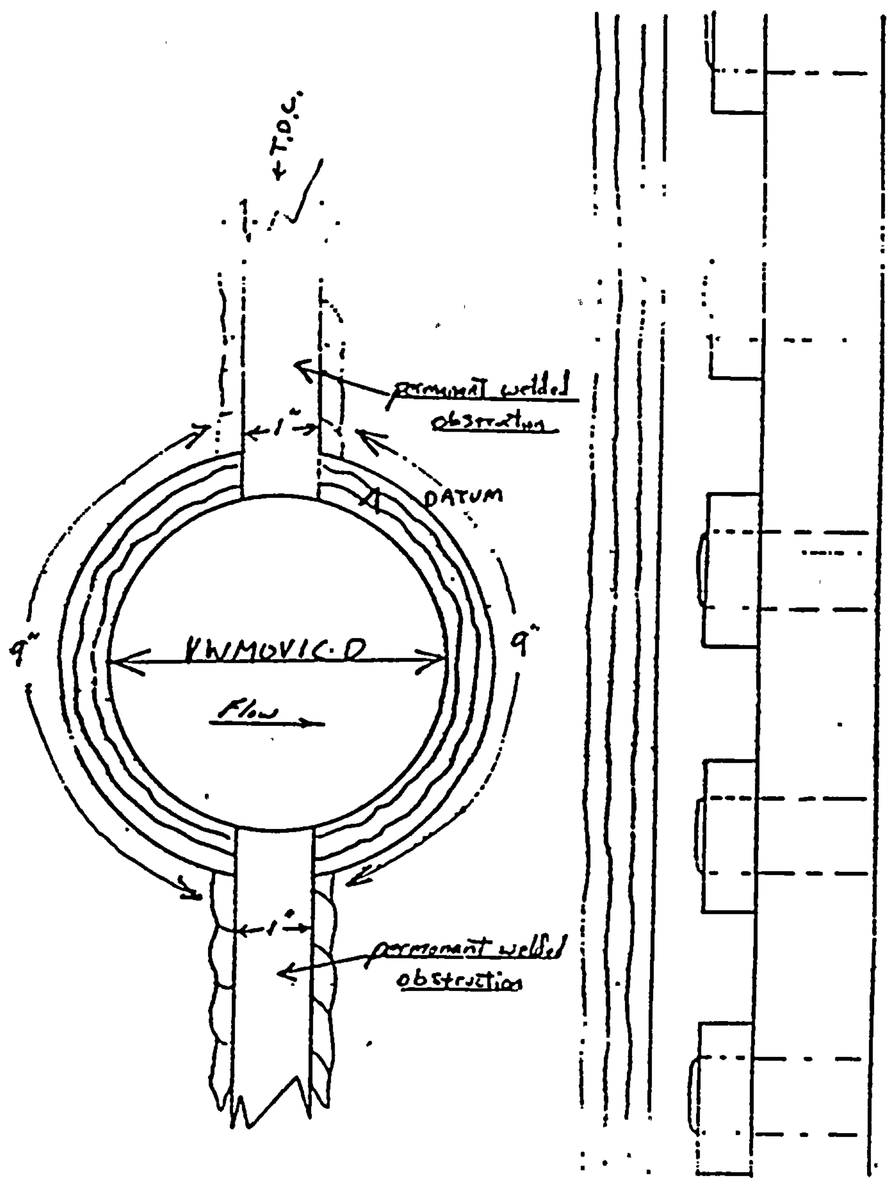
SKETCH SHEET

Area/System R.H.S. / ISI 66-22 Rev 1 / ISI Sketch 101 Page 2 of 2

Data Sheet No. ~~9920-1459A~~

Item No. 2-RHS-66-22-VWMOVIC-D



INFORMATION ONLY

Examiner [signature] Level II Date 1-13-86

Examiner N/A Level N/A Date N/A

REVIEWER [signature] LEVEL III DATE 1-24-86

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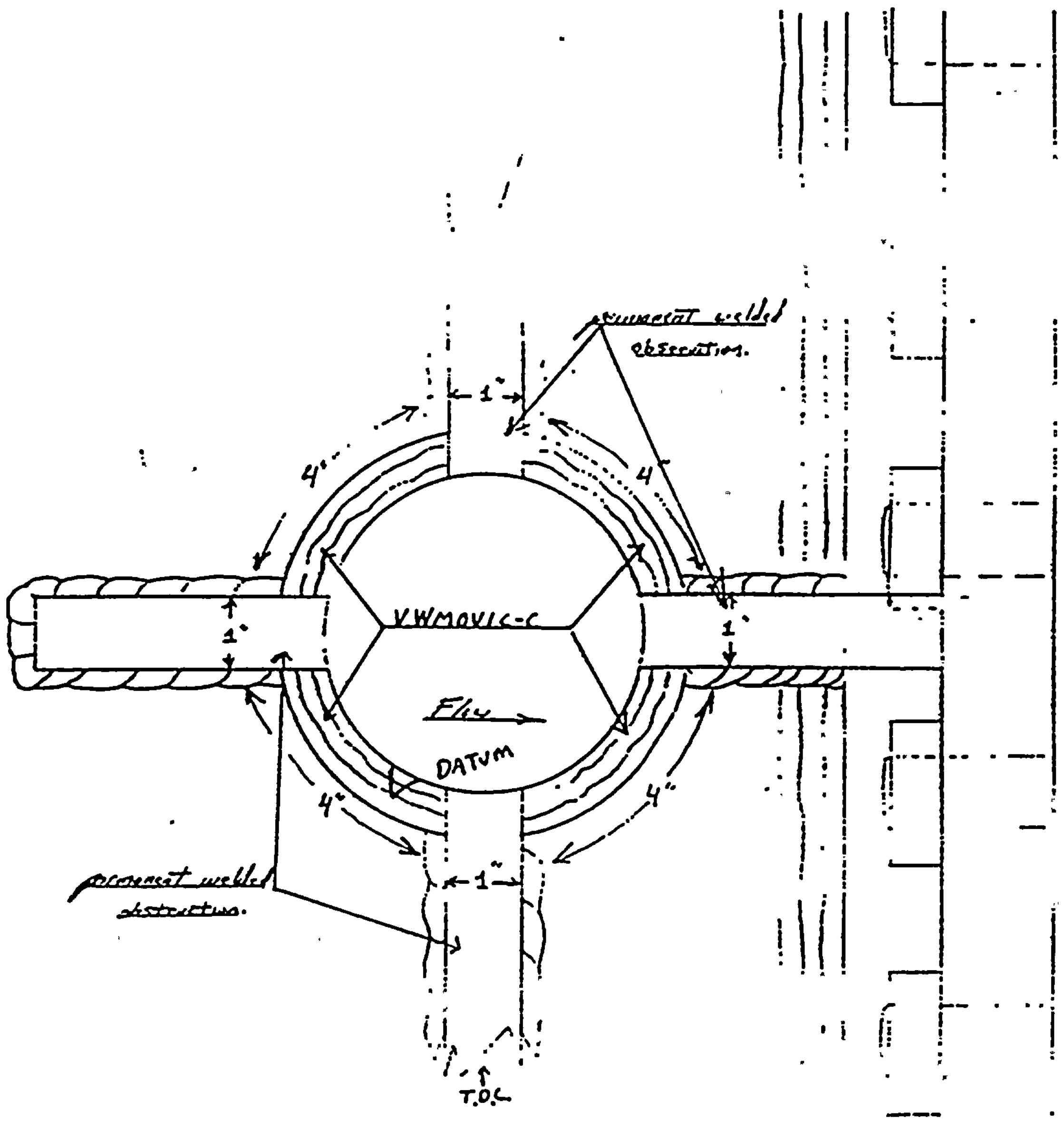
SKETCH SHEET

Area/System RHS/ISI-66-22 Rev 1/ISI Sketch #101

Data Sheet No. ~~9930-1868~~ 9931-163

Item No. 2RHS-66-22-VWMOVIC-C



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Examiner Andre W Moore Level II Date 1-13-86

Examiner N/A Level N/A Date N/A

REVIEWER [signature] LEVEL III DATE 1-24-86

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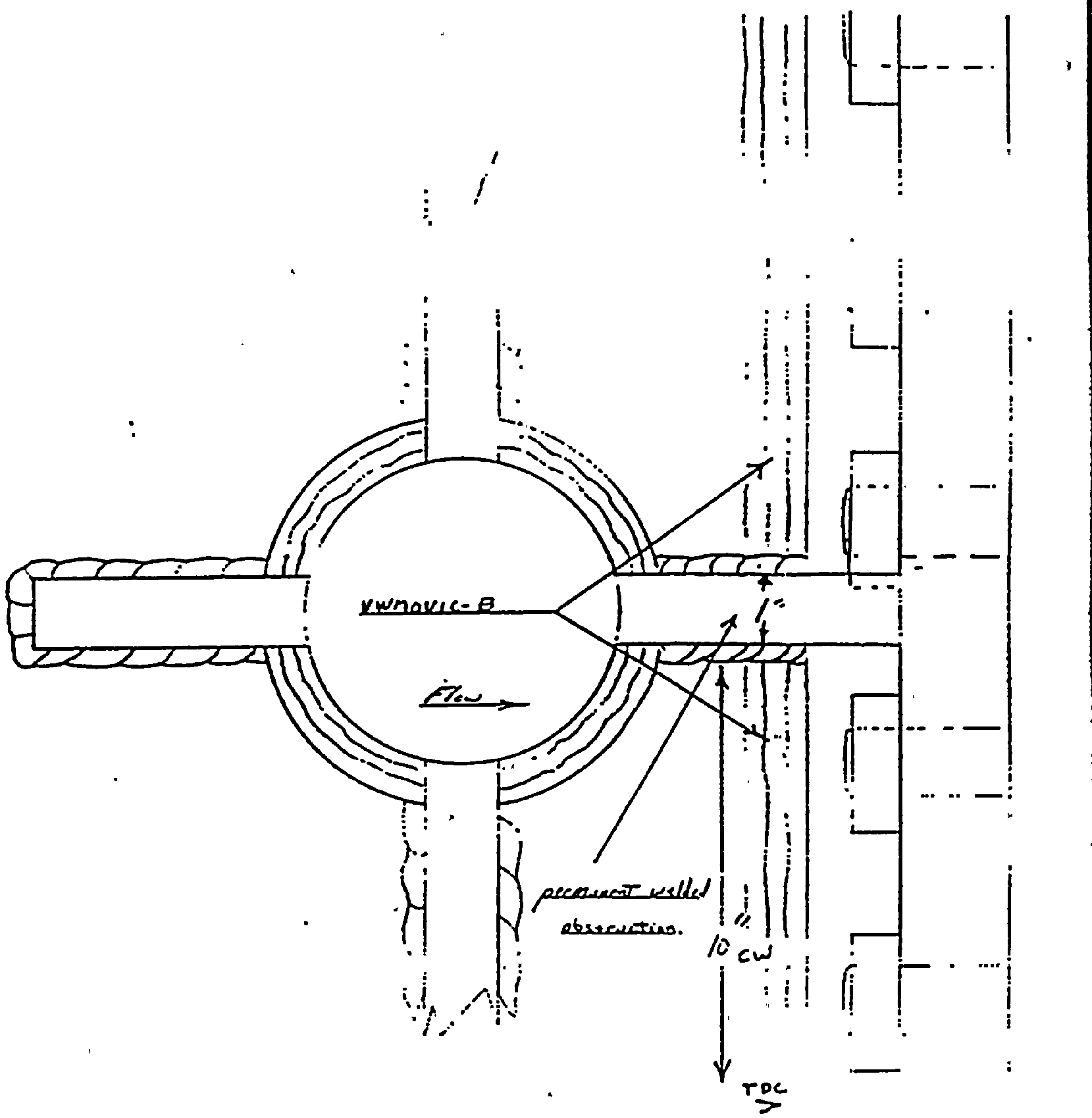
SKETCH SHEET

Area/System RHS / ISI 66-23 Rev 1 / ISI Sketch #101 Page 2 of 3

Data Sheet No 9920-1455A

Item No. 2-RHS-66-23-VWMOVIC-B



INFORMATION ONLY

Examiner Andrew W Morris Level II Date 1-13-86

Examiner N/A Level N/A Date N/A

REVIEWER [signature] LEVEL III DATE 1-24-86

NES NUCLEAR ENERGY SERVICES, INC.

ATTACHMENT 2

REVISIONS TO ORIGINAL RELIEF REQUESTS

Nine Mile Point Unit 2

Relief Request No.: RR-IWC-1

RELIEF REQUEST FOR RHR, HPCS, AND
LPCS SYSTEM PUMP CASING WELDS

1. Identification of Components

   Components - 2RHS*P1A, P1B, P1C pump casing welds
   - 2CSH*P1 pump casing welds
   - 2CSL*P1 pump casing welds

   See the attached ISI sketches:

   004A, 004B, 004C, 010A, 010B, 010C, 088A, 088B, 089A, 089B, 090A, and 090B, which indicate the welds for which relief is requested.

2. ASME Section XI Requirements

   Pump casing welds require a surface examination in accordance with Table IWC-2500-1, Category C-G, Item No. C6.10.

3. Basis for Relief

   The pump casing welds listed are inaccessible due to the pump being installed in a concrete pit, which does not allow access to the pump casing exterior to perform surface examinations. The pumps would require disassembly in order to perform the required examination. Pump disassembly at this stage of construction is considered impractical and not in the interest of safety from both a potential pump damage and disassembly/reassembly error.

   In addition, the structural integrity of the pump(s) pressure boundary(s) has been demonstrated during fabrication and construction and will undergo hydrostatic testing as part of ASME Section III final certification.

4. Inspection Period for Relief Request

   Preservice Inspection.

5. Alternate Tests or Examinations

   The ASME Section III nondestructive examinations indicated on pages 3 and 4 of 4 will be used. In addition, the pumps will receive further testing

in accordance with ASME Section XI, Article IWP, which further ensures the pumps' structural integrity.

6. Schedule for Implementing Alternate Test

   The alternate test will be performed during PSI.

7. Impact to Overall Plant Level of Quality

   No impact.

8. Preservice Examination Results

   The results of the ASME Section III examinations indicated on pages 3 and 4 of 4 will be submitted in the Final Summary Report.

9. Radiation Considerations

   None.

WELD NUMBER	COMPONENT/ WELD NAME	EXAM CAT	ITEM NO.	EXAM REQ	ASME III EXAM METHOD
PW207	2CSH*P1 Item 2.2 to 2.1	C-G	C6.10	Sur	Vol
PW208	2CSH*P1 Item 2.2 to 2.2	C-G	C6.10	Sur	Vol
PW209	2CSH*P1 Item 2.2 to 2.6	C-G	C6.10	Sur	Vol
PW212	2CSH*P1 Item 2.2 to 2.2	C-G	C6.10	Sur	Vol
PW217	2CSH*P1 Item 2.2 Long Weld	C-G	C6.10	Sur	Vol
PW218	2CSH*P1 Item 2.2 Long Weld	C-G	C6.10	Sur	Vol
PW219	2CSH*P1 Item 2.2 Long Weld	C-G	C6.10	Sur	Vol
PW311	2CSL*P1 Item 2.3 to 2.1	C-G	C6.10	Sur	Vol
PW312	2CSL*P1 Item 2.3 to 2.5	C-G	C6.10	Sur	Vol
PW315	2CSL*P1 Item 2.3 Long Weld	C-G	C6.10	Sur	Vol
PW111A	2RHS*P1A Item 2.3 to 2.1	C-G	C6.10	Sur	Vol
PW111B	2RHS*P1B Item 2.3 to 2.1	C-G	C6.10	Sur	Vol
PW111C	2RHS*P1C Item 2.3 to 2.1	C-G	C6.10	Sur	Vol
PW112A	2RHS*P1A Item 2.3 to 2.3	C-G	C6.10	Sur	Vol

WELD NUMBER	COMPONENT/ WELD NAME	EXAM CAT	ITEM NO.	EXAM REQ	ASME III EXAM METHOD
PW112B	2RHS*P1B Item 2.3 to 2.3	C-G	C6.10	Sur	Vol
PW112C	2RHS*P1C Item 2.3 to 2.3	C-G	C6.10	Sur	Vol
PW113A	2RHS*P1A Item 2.3 to 2.5	C-G	C6.10	Sur	Vol
PW113B	2RHS*P1B Item 2.3 to 2.5	C-G	C6.10	Sur	Vol
PW113C	2RHS*P1C Item 2.3 to 25.	C-G	C6.10	Sur	Vol
PW116A	2RHS*P1A Item 2.3 Upper	C-G	C6.10	Sur	Vol
PW116B	2RHS*P1B Item 2.3 Upper	C-G	C6.10	Sur	Vol
PW116C	2RHS*P1C Item 2.3 Upper	C-G	C6.10	Sur	Vol
PW118A	2RHS*P1A Item 2.3 Lower	C-G	C6.10	Sur	Vol
PW118B	2RHS*P1B Item 2.3 Lower	C-G	C6.10	Sur	Vol
PW118C	2RHS*P1C Item 2.3 Lower	C-G	C6.10	Sur	Vol

Nine Mile Point Unit 2

Relief Request No.: RR-IWC-2

RELIEF REQUEST FOR ASME SECTION III CLASS 2 PIPING
SUBMERGED UNDERWATER OR OTHERWISE INACCESSIBLE

1. Identification of Components

   Pages 2 of 3 through 3 of 3 identify certain welds in the CSH, CSL, ICS and RHS systems classified in accordance with ASME Section XI Examination Category C-F, which are submerged in the suppression pool or otherwise inaccessible.

2. ASME Section XI Requirements

   Piping is nonexempt and requires a volumetric and/or surface examination in accordance with Table IWC-2500-1, Category C-F, Item No. C5.11 and/or C5.21.

3. Basis for Relief

   Relief is requested due to inaccessibility of the welds located underwater within the suppression pool or otherwise inaccessible.

   Structural integrity of those portions of the piping system has been demonstrated during fabrication and erection under ASME Section III. The piping has been designed for submerged conditions and postulated plant loading combinations. Postulated cracks in this piping are not detrimental to the safety function of the systems in which these lines are located.

4. Inspection Period for Relief Request

   Preservice Inspection.

5. Alternate Tests or Examinations

   These welds received RT exam for ASME III

6. Schedule for Implementing Alternate Tests

   Prior to completion of PSI.

7. Impact to Overall Plant Level of Quality

   No impact.

8. Preservice Examination Results

   ASME Section III RT examination results will be submitted in the Final Summary Report.

9. Radiation Considerations

   None.

Nine Mile Point Unit 2

RR-IWC-2 (cont)

WELD NUMBER	COMPONENT/ WELD NAME	EXAM CAT	ITEM NO.	EXAM REQ
25-05-CSH-FW012	Pipe/Elbow	C-F	C5.11	Sur
25-05-CSH-FW013	Pipe/WNF	C-F	C5.11	Sur
25-05-CSH-FW014	Elbow/Z-12	C-F	C5.11	Sur
25-19-CSH-FW011*	Elbow/Z-13	C-F	C5.11	Sur
25-19-CSH-SW013*	Pipe/Elbow	C-F	C5.11	Sur
26-01-CSL-FW026	Pipe/Elbow	C-F	C5.11	Sur
26-01-CSL-FW027	Pipe/WNF	C-F	C5.11	Sur
26-01-CSL-FW028	Elbow/Z-15	C-F	C5.11	Sur
26-01-CSL-FW035	Pipe/Pipe	C-F	C5.11	Sur
57-08-ICS-FW007*	Pipe/Pene Z-19	C-F	C5.11	Sur
57-08-ICS-FW015*	Pipe/Elbow	C-F	C5.11	Sur
57-08-ICS-FW016*	Pipe/Elbow	C-F	C5.11	Sur
57-08-ICS-SW056	Pipe/Plate	C-F	C5.11	Sur
Diff 1-A*	Pipe/Z-88A	C-F	C5.21	Vol and Sur
Diff 1-B*	Pipe/Pipe	C-F	C5.21	Vol and Sur
Diff 1-C*	Pipe/Pipe	C-F	C5.21	Vol and Sur
Diff 1-D*	Pipe/Pipe	C-F	C5.21	Vol and Sur
Diff 1-E	Pipe/Flange	C-F	C5.21	Vol and Sur
Diff 1-M	Pipe/Flange	C-F	C5.21	Vol and Sur
Diff 2-A*	Pipe/Z-88B	C-F	C5.21	Vol and Sur
Diff 2-B*	Pipe/Pipe	C-F	C5.21	Vol and Sur
Diff 2-C*	Pipe/Pipe	C-F	C5.21	Vol and Sur
Diff 2-D*	Pipe/Pipe	C-F	C5.21	Vol and Sur
Diff 2-E	Pipe/Flange	C-F	C5.21	Vol and Sur
Diff 2-M	Pipe/Flange	C-F	C5.21	Vol and Sur
66-08-RHS-FW001	Pipe/Pipe	C-F	C5.11	Sur
66-08-RHS-FW003*	Pipe/Z73	C-F	C5.11	Sur
66-08-RHS-FW011*	Pipe/Elbow	C-F	C5.11	Sur

*Indicates weld is not submerged, but is otherwise inaccessible

WELD NUMBER	COMPONENT/ WELD NAME	EXAM CAT	ITEM NO.	EXAM REQ
66-08-RHS-SW018*	Pipe/Elbow	C-F	C5.11	Sur
66-13-RHS-FW023	Pipe/Elbow	C-F	C5.11	Sur
66-13-RHS-FW024	Pipe/WNF	C-F	C5.11	Sur
66-13-RHS-FW025	Z5A/Elbow	C-F	C5.11	Sur
66-13-RHS-FW029	Pipe/Pipe	C-F	C5.11	Sur
66-17-RHS-FW010*	Pipe/Z6B	C-F	C5.11	Sur
66-17-RHS-SW016*	Pipe/Elbow	C-F	C5.11	Sur
66-22-RHS-FW021	Pipe/Elbow	C-F	C5.11	Sur
66-22-RHS-FW022	Pipe/WNF	C-F	C5.11	Sur
66-22-RHS-FW023	Pipe/Z5C	C-F	C5.11	Sur
66-22-RHS-FW029	Pipe/Pipe	C-F	C5.11	Sur
66-23-RHS-FW018	Flg/Pipe	C-F	C5.11	Sur
66-23-RHS-FW019	Pipe/Sr Elb	C-F	C5.11	Sur
66-23-RHS-FW020	Elb/Z-5B	C-F	C5.11	Sur
66-23-RHS-FW022	Pipe/Pipe	C-F	C5.11	Sur
66-28-RHS-FW007*	Z-6A/Elbow	C-F	C5.11	Sur
66-28-RHS-SW006*	Elb/Pipe	C-F	C5.11	Sur

*Indicates weld is not submerged, but is otherwise inaccessible

Nine Mile Point Unit 2

Relief Request No.: RR-IWC-3

RELIEF REQUEST FOR ASME SECTION III, CLASS 2 DIFFUSERS (2RHS*DIFF1 and DIFF2) SHELL WELDS

1. Identification of Components

   Page 3 of 3 identifies those welds as shown on ISI Isometric Drawing Nos. ISI-66-11 and 66-07 for which relief is requested.

2. ASME Section XI Requirements

   Volumetric examination is required in accordance with Table IWC-2500-1, Category C-A, Item No. C1.10.

3. Basis for Relief

   Relief is requested due to inaccessibility of welds located underwater within the suppression pool.

   Structural integrity of the diffusers has been demonstrated during fabrication and installation under ASME Section III. A through-wall crack in the diffuser is not detrimental to the safety function of the RHS system.

4. Inspection Period for Relief Request

   Preservice Inspection

5. Alternate Tests or Examinations

   These welds received ASME Section III RT examination.

6. Schedule for Implementing Alternate Tests

   Prior to completion of PSI

7. Impact to Overall Plant Level of Quality

   No impact

8. Preservice Examination Results

   ASME Section III RT examination results will be submitted in the Final Summary Report.

Nine Mile Point Unit 2

Relief Request No.: RR-IWC-4

RELIEF REQUEST FOR ASME SECTION III, CLASS 2
DIFFUSERS (2RHS*DIFF1 and DIFF2)
NOZZLE-TO-SHELL WELDS

1. Identification of Components

   Page 3 of 3 identifies those welds as shown on ISI Isometric Drawing Nos. ISI-66-11 and 66-07 for which relief is requested.

2. ASME Section XI Requirements

   Volumetric and surface examinations are required in accordance with Table IWC-2500-1, Category C-B, Item No. C2.21.

3. Basis for Relief

   Relief is requested due to the inaccessibility of welds located underwater within the suppression pool.

   Structural integrity of the diffusers has been demonstrated during fabrication and installation under ASME Section III. A through-wall crack in the diffuser is not detrimental to the safety function of the RHS system.

4. Inspection Period for Relief Request

   Preservice Inspection

5. Alternate Tests or Examinations

   These welds received ASME Section III RT examination.

6. Schedule for Implementing Alternate Tests

   Prior to completion of PSI

7. Impact to Overall Plant Level of Quality

   No impact

8. Preservice Examination Results

   ASME Section III RT examination results will be submitted in the Final Summary Report.

9. Radiation Considerations

   None

Relief Request No.: RR-IWC-5

RELIEF REQUEST FOR ASME SECTION III, CLASS 2 PIPING INTEGRAL ATTACHMENTS

1. Identification of Components

   Page 3 of 3 identifies piping integral attachment welds.

2. ASME Section XI Requirements

   Piping integral attachment welds require a surface examination in accordance with Table IWC-2500-1, Category C-C, Item No. C3.40.

3. Basis for Relief

   Relief is requested due to the inaccessibility of the welds because they are submerged.

   Structural integrity of those portions of the piping boundary has been demonstrated during the fabrication and installation under ASME Section III. The piping and the related integral attachments have been designed for submerged conditions and postulated plant loading combinations. Postulated cracks in this piping are not detrimental to the safety function of the CSL system.

4. Inspection Period for Relief Request

   Preservice Inspection

5. Alternate Tests or Examinations

   These welds received ASME Section III surface examination.

6. Schedule for Implementing Alternate Tests

   Prior to completion of PSI

7. Impact to Overall Plant Level of Quality

   No impact

8. Preservice Examination Results

   ASME Section III surface examination results shall be submitted in the Final Summary Report.

9. Radiation Considerations

None

Nine Mile Point Unit 2

Relief Request No.: RR-IWC-6

RELIEF REQUEST FOR RHR,
HPCS, AND LPCS SYSTEM (ASME CLASS 2)
PUMP INTEGRAL ATTACHMENTS

1. Identification of Components

   Pages 3 of 5 through 5 of 5 list pump integral attachments in the CSH, CSL, and RHS pumps classified in accordance with ASME Section XI Examination Category C-C. Refer to the pump drawings with Relief Request RR-IWC-1 which identifies the pump weld numbers.

2. ASME Section XI Requirements

   Pump integral attachments require a surface examination in accordance with Table IWC-2500-1, Category C-C, Item No. C3.70.

3. Basis for Relief

   The pump integral attachments are inaccessible due to the pump being installed in a concrete pit, which does not allow access to the pump casing to perform surface examinations. The pumps would require disassembly to perform the required examination. Pump disassembly at this stage of construction is considered impractical and not in the interest of safety from both a potential pump damage and disassembly/reassembly error.

   In addition, the structural integrity of the pump(s) pressure boundary(s) has been demonstrated during fabrication and construction and will undergo hydrostatic testing as part of the ASME Section III final certification.

4. Inspection Period for Relief Request

   Preservice Inspection.

5. Alternate Tests or Examinations

   The ASME Section III nondestructive examinations indicated on pages 3, 4 and 5 of 5 will be used. In addition, the pumps will receive further testing in accordance with ASME Section XI, Article IWP, which further ensures the pumps' structural integrity.

6. Schedule for Implementing Alternate Test

   The alternate test will be performed during PSI.

7. Impact to Overall Plant Level of Quality

   No impact.

8. Preservice Examination Results

   The results of the ASME Section III examinations indicated on pages 3, 4 and 5 of 5 will be submitted in the Final Summary Report.

9. Radiation Considerations

   None.

WELD NUMBER	COMPONENT/ WELD NAME	EXAM CAT	ITEM NO.	EXAM REQ	ASME III EXAM METHOD
PW202	2CSH*P1 Item 1.4 to 1.7	C-C	C3.70	Sur	Sur
PW211	2CSH*P1 Item 1.14 to 1.2	C-C	C3.70	Sur	Sur
PW220	2CSH*P1 Item 2.5 to 2.2	C-C	C3.70	Sur	Sur
PW221	2CSH*P1 Item 2.5 to 2.2	C-C	C3.70	Sur	Sur
PW222	2CSH*P1 Item 2.7 to 2.2	C-C	C3.70	Sur	Sur
PW223	2CSH*P1 Item 2.7 to 2.2	C-C	C3.70	Sur	Sur
PW224	2CSH*P1 Item 2.4 to 2.2	C-C	C3.70	Sur	Sur
PW225	2CSH*P1 Item 2.4 to 2.2	C-C	C3.70	Sur	Sur
PW226	2CSH*P1 Item 2.3 to 2.2	C-C	C3.70	Sur	Sur
PW227	2CSH*P1 Item 2.3 to 2.2	C-C	C3.70	Sur	Sur
PW229	2CSH*P1 Item 1.9 to 1.7	C-C	C3.70	Sur	Sur
PW230	2CSH*P1 Item 1.9 to 1.7	C-C	C3.70	Sur	Sur
PW307	2CSL*P1 Item 1.15 to 1.2	C-C	C3.70	Sur	Sur
PW319	2CSL*P1 Item 2.2 to 2.5	C-C	C3.70	Sur	Sur
PW323	2CSL*P1 Item 2.4 to 2.3	C-C	C3.70	Sur	Sur

WELD NUMBER	COMPONENT/ WELD NAME	EXAM CAT	ITEM NO.	EXAM REQ	ASME III EXAM METHOD
PW105A	2RHS*P1A Item 1.15 to 1.9	C-C	C3.70	Sur	Sur
PW105B	2RHS*P1B Item 1.15 to 1.9	C-C	C3.70	Sur	Sur
PW105C	2RHS*P1C Item 1.15 to 1.9	C-C	C3.70	Sur	Sur
PW107A	2RHS*P1A Item 1.15 to 1.2	C-C	C3.70	Sur	Sur
PW107B	2RHS*P1B Item 1.15 to 1.2	C-C	C3.70	Sur	Sur
PW107C	2RHS*P1C Item 1.15 to 1.2	C-C	C3.70	Sur	Sur
PW121A	2RHS*P1A Item 2.2 to 2.5	C-C	C3.70	Sur	Sur
PW121B	2RHS*P1B Item 2.2 to 2.5	C-C	C3.70	Sur	Sur
PW121C	2RHS*P1C Item 2.2 to 2.5	C-C	C3.70	Sur	Sur
PW125A	2RHS*P1A Item 1.13 to 1.1	C-C	C3.70	Sur	Sur
PW125B	2RHS*P1B Item 1.13 to 1.1	C-C	C3.70	Sur	Sur
PW125C	2RHS*P1C Item 1.13 to 1.1	C-C	C3.70	Sur	Sur
PW126A	2RHS*P1A Item 1.13 to 1.1	C-C	C3.70	Sur	Sur
PW126B	2RHS*P1B Item 1.13 to 1.1	C-C	C3.70	Sur	Sur
PW126C	2RHS*P1C Item 1.13 to 1.1	C-C	C3.70	Sur	Sur

RR-IWC-6 (Cont)

WELD NUMBER	COMPONENT/ WELD NAME	EXAM CAT	ITEM NO.	EXAM REQ	ASME III EXAM METHOD
PW127A	2RHS*P1A Item 2.4 to 2.3	C-C	C3.70	Sur	Sur
PW127B	2RHS*P1B Item 2.4 to 2.3	C-C	C3.70	Sur	Sur
PW127C	2RHS*P1C Item 2.4 to 2.3	C-C	C3.70	Sur	Sur
PW128A	2RHS*P1A Item 2.4 to 2.3	C-C	C3.70	Sur	Sur
PW128B	2RHS*P1B Item 2.4 to 2.3	C-C	C3.70	Sur	Sur
PW128C	2RHS*P1C Item 2.4 to 2.3	C-C	C3.70	Sur	Sur

Enclosure 1

Summary of Items Discussed

From GL 83-28 for NMP-2

ITEM

1.1 NMPC should revise the response to GL 83-28, item 1.1 and Administrative Procedure (AP) - 4, 7.4.2 to state that "not understood" conditions should be reviewed by SORC. The applicant agreed to make that revision.

1.2 Post Trip

For the sequence of events and analog time history, NMPC needs to respond to these parameters specifically and state that they are included.

1.2.1.5 and 1.2.2.5 - AP 10.1 should state it is for the life of the plant.

1.2.2.2 - NMPC needs to include the sampling rate for the analog.

1.2.2.3 - The duration for the time history should be at least 5 min. pre-trip to 10 min. post-trip. (NMPC stated the duration is 5 min. pre-trip to 5 min. post-trip for NSSS and 5 min. pre-trip to 30 min. post-trip for BOP.) If NSSS parameters are needed then the duration for these parameters should be extended. If these parameters are on strip chart recorders powered by a UPS then NMP-2 may have continual sampling covering the 10 min. post-trip anyway.

1.2.4 - If SPDA is not needed to cover requirements of GL 83-28 then NMPC may wish to delete the reference to it.

2.1 and 2.2 2 categories

a) Environmental Qualifications (EQ).

Q-list is being subdivided into Master EQ List (MEL)

b) Vendor Interface

The GE SIL program and NMPC's effort to keep updated should be discussed. NMPC should expand on the GE Operations Engineer Program. The transition process to an operating license should be discussed.

2.2.1.1 - NMPC needs to provide a brief description of the process for determining if a component needs to be safety class or not.

2.2.1.1 and 2.2.1.4 - See enclosure 3 to the meeting summary.

2.2.1.5 - NMPC's draft response referenced the NRC EQ audit to demonstrate their program. The NRC stated that NRC audits do not demonstrate NMPC programs and an NMPC demonstration is needed here. NMPC was told to include a program both for what is in the plant now and for replacement parts.

2.2.2 - See enclosure 3 to the meeting summary.

3.1 and 3.2 3.1.1 and 3.2.1 - NMPC needs to provide additional information concerning the program.

<u>3.1.2 and 3.2.2</u>

a) Has NMPC reviewed all vendor recommendations?

b) Has all appropriate information been incorporated?

NMPC will provide a statement on what has been done and what will be done to complete these issues and a schedule for implementation. NMPC should include other vendors in addition to General Electric.

4.5.1 SCRAM pilot valves are tested once a week on a ½ SCRAM. Backup SCRAM valves should be tested once every 18 months.

4.5.3 NMPC will include a statement that NMP-2 was part of the GE Owners Group.

Enclosure 2
DRAFT

Preface

Throughout this document are references to NMPC or SWEC procedures. Except as specified in the response, these procedures are attached to this letter (listed below) and are included to facilitate NRC's review of this document. These procedures are, and must be "living documents" that will undoubtedly be revised in the future. Their inclusion here does not constitute any commitment by NMPC or SWEC to maintain these procedures verbatim as presented here. However, NMPC does commit to maintaining compliance with the intent of Generic Letter 83-28 as specified herein.

Attachment #1	AP-1.1	Composition and Responsibility of Site Organization
Attachment #2	AP-1.2	Composition and Responsibility of Unit Organization
Attachment #3	AP-1.3	Personnel Responsibilities and Authority
Attachment #4	AP-2	Production and Control of Procedures
Attachment #5	AP-3.4.1	Administration of Technical and Safety Reviews - Site Operations Review Committee
Attachment #6	AP-3.4.2	Operations Experience Assessment
Attachment #7	AP-4.0	Administration of Operations
Attachment #8	AP-5.0	Procedure for Repair
Attachment #9	AP-10.1	Management of Station Records
Attachment #10	TDP-5	Administration of Operational Engineering Assessment Items
Attachment #11	TDP-6	Nuclear Plant Reliability Data System (NPRDS) Failure Reporting
Attachment #12	TDP-8	Post-Maintenance Testing Criteria
Attachment #13	TDP-9	Independent Safety Engineering Group
Attachment #14	NTP-10	Training & Licensed Operator Candidates
Attachment #15	NTP-11	Licensed Operator Retraining
Attachment #16	N2-IOP-101A	Plant Startup
Attachment #17	N2-RAP-6	Post Reactor Scram Analysis and Evaluation
Attachment #18	SWEC Procedure PP-81	Method for Handling Supplier Technical Documents

Attachment #19	NEL-014.G	Control and Distribution of Vendors Documents
Attachment #20	SWEC Procedure C-3	Equipment Identification Codes

Section 1.1

Generic Letter 83-28

Post-Trip Review (Program Description and Procedure)

REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS OF SALEM ATWS EVENTS

1.1 POST-TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)

Position

Licensees and applicants shall describe their program for ensuring that unscheduled reactor shutdowns are analyzed and that a determination is made that the plant can be restarted safely. A report describing the program for review and analysis of such unscheduled reactor shutdowns should include, as a minimum:

1.1.1 The criteria for determining the acceptability of restart.

NMP2 Response

Nine Mile Point Unit 2's criteria for determining the acceptability of restart are contained in Procedure N2-RAP-6, Post Reactor Scram Analysis and Evaluation, and in (Interim) Operating Procedure N2-IOP-101A, Plant Startup (IOP-101A will be approved as OP-101A prior to startup). N2-RAP-6 provides a review and evaluation of specific parameters associated with a Reactor Scram from all operating conditions. If after the completion of this procedure, there is a condition which is not fully understood, The Site Operations Review Committee (SORC) must review this report before the Station Superintendent can authorize a restart. In the operating procedure N2-IOP-101A, valve instrumentation, system and component checkoff sheets must be completed prior to reactor startup. These pre-startup checkoff sheets are used to ensure that all equipment, necessary for safe operation is operable in accordance with plant Technical Specifications. This procedure also states that N2-RAP-6 must be completed (following a Scram) prior to restart.

The Administrative Procedure which identifies the criteria that the Station Superintendent will use for determining the acceptability of restart is, AP-4, Administration of Operations. Section 7.4 of this procedure states as follows:

7.4 The criteria in which the Station Superintendent will use for determining the acceptability of restart, after an unscheduled shutdown, shall be as follows:

7.4.1 The plant is shown to be in a safe condition.

7.4.2 The cause of the event is either understood or, after a detailed investigation, is considered to have been a spurious trip with a reasonably low potential for reoccurrence.

1.1.1 (Cont'd)

7.4.3 The need for corrective action has been determined and appropriately implemented.

7.4.4 The expected automatic operation of plant safety related systems has been verified.

Therefore, Unit 2 is currently in compliance with the intent of Section 1.1.1.

1.1.2 The responsibilities and authorities of personnel who will perform the review and analysis of these events.

NMP2 Response

The Superintendent Operations, Station Shift Supervisor, Shift Technical Advisor, and Technical Department personnel will perform the Post-Trip Review analysis. Their duties are specifically stated in Administrative Procedures AP-1.2 and AP-1.3, Composition and Responsibility of Unit Organization and Personnel Responsibilities and Authority. They are assisted by the Reactor Analyst Department which is directly responsible for the completion of N2-RAP-6, Post Reactor Scram Analysis and Evaluation Procedure. This procedure states specifically that "the Reactor Analyst Department will be directly responsible for data gathering and process evaluation. The analysis will be completed by the Unit Reactor Analyst or Site Reactor Analyst. In the event that those individuals are unavailable, the analysis will be conducted by a senior member of Technical Services and/or operations". Their duties are also supported by Site Administrative Procedure AP-1.1, Composition and Responsibility of Site Organization.

Therefore, administrative controls which regulate the responsibilities and authorities of personnel evaluating the Post-Trip Review meet the intent of Section 1.1.2.

1.1.3 The necessary qualifications and training for the responsible personnel.

NMP2 Response:

The analysis of unscheduled shutdowns at Nine Mile Point Unit 2 will be performed by a select group of trained and qualified individuals. The individuals currently in the positions of Superintendent Operations, Site Reactor Analyst, Unit Reactor Analyst, and the Station Shift Supervisors all have experience at the operating facilities at Nine Mile Point Unit 1 and/or James A. Fitzpatrick. The education, training, and job related experience qualify these people to make the Post-Trip Review and restart recommendation.

1.1.3 (Cont'd)

The qualifications and training for the positions of Station Superintendent, Superintendent Operations, Site Reactor Analyst Supervisor and Unit Reactor Analyst Supervisor comply with the requirements of ANSI/ANS 3.1-1978. Additionally, the current site Reactor Analyst Supervisor and the Unit Reactor Analyst Supervisor both hold Senior Reactor Operator licenses at Unit 2. The Station Shift Supervisors and Shift Technical Advisors also will meet the qualification requirements of ANSI/ANS 3.1-1978. The Shift Technical Advisor meets the Commission's Policy Statement on engineering expertise described in 50FR43621.

The training procedures which Unit 2 utilizes are NTP-10, Training of Licensed Operator Candidates and NTP-11, Licensed Operator Retraining. These procedures formally establish the procedures, programs, responsibilities and requirements necessary for the qualifications of NRC Licensed Reactor Operators and Senior Operators at Nine Mile Point 2.

Therefore, the existing Nine Mile Point Unit 2 administrative controls currently meet the intent of Section 1.1.3.

1.1.4 The sources of plant information necessary to conduct the review and analysis. The sources of information should include the measures and equipment that provide the necessary detail and type of information to reconstruct the event accurately and in sufficient detail for proper understanding. (See Action 1.2)

NMP2 Response

Information necessary to conduct the review and analysis is available, to the responsible personnel, through a number of different sources. The main source of data will come from N2-RAP-6, Post Reactor Scram Analysis and Evaluation Procedure. This procedure is designed to evaluate system performance from an initiation or isolation standpoint. The determination of safety system initiation, proper flow paths and system operation will be done using post trip logs, control room instrumentation, recorders, alarms, indicating lights, and the General Electric Transient Analysis Recording System (GETARS), as well as the Unit 2 Process Computer System. These systems provide Operators with essential plant performance information through a variety of logs, trends, summaries, and data displays. More information on these systems is provided in section 1.2 (Post-Trip Review - Data and Information Capability).

1.1.5 The methods and criteria for comparing the event information with known or expected plant behavior (e.g., that safety-related equipment operates as required by the Technical Specifications or other performance specifications related to the safety function).

1.1.5 (Cont'd)

NMP2 Response

As stated in Section 1.1.3, the individuals responsible for the event analysis are qualified per ANSI/ANS 3.1-1978 and currently hold Senior Reactor Operator licenses (both Unit and Site Reactor Analysts). At their disposal are records of previous reactor trips (when history records exist), Technical Specifications, Final Safety Analysis Report data, and reload licensing analyses which are used at their discretion for comparing the transient to expected responses.

1.1.6 The criteria for determining the need for independent assessment of an event (e.g., a case in which the cause of the event cannot be positively identified, a competent group such as the Site Operations Review Committee, will be consulted prior to authorizing restart) and guidelines on the preservation of physical evidence (both hardware and software) to support independent analysis of the event.

NMP2 Response

Unit 2's criteria for determining the need for independent assessment is contained in Reactor Analysis Procedure N2-RAP-6, Post Reactor Scram Analysis and Evaluation. This procedure specifically states (on the Final Assessment Sheet) that "If there is a condition not fully understood, the Station Superintendent should be so notified and the appropriate staff members called in to assist in the evaluation. If after further evaluation the scram is still not understood, SORC must review this report before authorization to restart". Also, AP-3.4.1, Administration of Technical and Safety Reviews, (SORC) states: "Scram reports need not be reviewed by SORC prior to restart unless the cause of the scram or the plant transient response is not fully understood. Under these conditions SORC will provide the independent assessment per generic letter 83-28 Section 1.1.6, and SORC approval is required prior to restart". Section 1.1.1 (of this response) states specific criteria contained in Administrative Procedure AP-4 which the Station Superintendent must follow prior to authorizing a restart.

Unit 2's procedure established to assure that all physical evidence (necessary for an independent assessment) is preserved is AP-10.1, Management of Station Records. This procedure provides an outline for the collection, storage and maintenance of site records and technical information. This procedure states that all Scram Reports and Scram Analysis data (N2-RAP-6) remain in plant archives for the life of the plant. This enables operating personnel to compare event information with known or expected plant behavior at any time.

Therefore, the Administrative Controls provide a systematic method to determine the need for independent assessment and NMP2 meets the intent of Section 1.1.6.

1.1.7 Our systematic safety assessment procedures which addresses Section 1.1 Post-Trip Review, are as follows:

Site Administrative Procedures

AP-1.1 Composition and Responsibility of Site Organization
AP-1.2 Composition and Responsibility of Unit Organization
AP-1.3 Personnel Responsibilities and Authority
AP-3.4.1 Administration of Technical and Safety Reviews (SORC)
AP-4.0 Administration of Operations
AP-10.1 Management of Station Records

Nuclear Training Procedures

NTP-10 Training of Licensed Operator Candidates
NTP-11 Licensed Operator Retraining

Reactor Analyst Procedure

N2-RAP-6 Post Reactor Scram Analysis and Evaluation

Operating Procedure

N2-IOP-101A Plant Startup

The administrative controls currently being implemented at Nine Mile Point Unit 2 contain procedures and data collection requirements related to Post-Trip Review. These requirements provide assurance that the cause for unscheduled reactor shutdown is analyzed and a determination made as to the cause prior to plant restart. In addition, the general response of safety related equipment is reviewed prior to plant restart.

Nine Mile Point Unit 2's Administrative Controls adequately addresses Sections 1.1 on Post-Trip Review.

Section 1.2

Generic Letter 83-28

Post-Trip Review (Data and Information Capability)

Section 1.2

Post-Trip Review - Data and Information Capability

Unit 2's Computer equipment which is capable of recording, recalling and displaying data and information necessary to diagnose the cause of unscheduled reactor shutdowns, is comprised of three different systems: The Process Computer System, General Electric's Transient Analysis Recording System, and the Safety Parameter Display System. Each system works independent of one another, but has many redundant data ID points which provide crucial information during a system failure.

The following three sections discuss each system in detail and answer the questions generated in Generic Letter 83-28.

<u>Section 1.2A</u>

Generic Letter 83-28

Post-Trip Review - Data and Information Capability

Process Computer System (PCS)

1.2.1 Capability for assessing sequency of events (on-off indications).

1.2.1.1 Brief description of equipment (e.g., plant computer, dedicated computer, strip chart).

NMP2 Response

The Process Computer installed at Nine Mile Point Unit 2 consists of dual Honeywell 4500 C.P.U.'s with the General Electric Process Management System (PMS) software package. Each processor contains 128K word memory for core storage and dual ported Ampex large core stores for bulk devices. In addition, the system utilizes an 80MB disk drive for additional storage capacity, and for back-up capability. Two magnetic tape units are utilized for either historical recording retention or for back-up capabilities.

For peripherals, the computer room is equipped with two color graphic videos, two input keyboards, two input/output terminets, one output only terminet, a cardreader, and a high speed line printer.

The control room is equipped with four color graphic videos, two input keyboards, one input/output terminet, two output terminets, six trend recorders, and five digital displays. Attachment A contains a list of all the main control room dedicated strip charts.

Additionally, the remote shutdown room is equipped with one color graphic video and one keyboard.

1.2.1.2 Parameters monitored.

NMP2 Response

Attachment 1 is a copy of all the sequence of event points that exist on the system to date. There has been a considerable amount of spares created so that points may be added in the future. These points reflect trip points associated with electrical breaker status, water levels, relief valve positions, IRM and APRM upscale levels, and the Neutron Monitoring System.

1.2.1.3 Time discrimination between events.

NMP2 Response

SOE (sequence of event) points are alarmed and recorded on an automatic interrupt driven basis on a change of state. Temporal resolution is 4 milliseconds between events. Events occurring within this time period may not be recorded in sequence.

1.2.1.4 Format for displaying data and information.

NMP2 Response

Attachment 2 is a copy of the format used when a sequence of event (SOE) log is printed out to a terminet. The log will print after recording 64 contact changes or 30 seconds after first contact change. Each change of state will also be alarmed to the alarm terminet in the Control Room. The time period to printout can be changed from 1-60 seconds.

1.2.1.5 Capability for retention of data and information.

NMP2 Response

Retention of all sequence of event (SOE) data is controlled by the Historical Recording System. The HRR (Historical Recording Retention) system will record all changes of state including SOE points. This data can then be retrieved at any time from either the disk drive or from magnetic tape depending on the time frame. Attachment 3 contains the data format viewed by the user. The distance back in time a user may go depends on the retention cycle of magnetic tapes used. The data can be printed to a terminet or viewed on the CRT screen.

1.2.1.6 Power source(s) (e.g., Class 1E, non-Class 1E, noninterruptible).

NMP2 Response

Power to the Unit 2 Process Computer is provided by an Uninterruptible Power Supply 2VBB-UPS1G Non-Class 1E. This supply is fed from a 600V power panel 2VBB-PNL301, which is supplied by either the Station Generator 13.8KV line, (2NJS-US3, during normal operation) or from an off-site Scriba 115KV line (2NJS-US4, during a shutdown condition). The process computer is also supplied by an alternate 600V bus 2NJS-US6. In the condition which all power is lost, backup power is supplied by a 125V DC battery supply, 2BYS-SWG001C.

In summary, upon loss of normal power, a static transfer switch transfers power from the normal source to the alternative source. If both normal and alternate sources are lost, the DC source will automatically pickup the loads by means of a DC auctioneering circuit.

1.2.2 Capability for assessing the time history of analog variables needed to determine the cause of unscheduled reactor shutdowns and the functioning of safety-related equipment.

1.2.2.1 Brief description of equipment (e.g., plant computer, dedicated computer, strip charts).

NMP2 Response

A brief description of the equipment comprising the Unit 2 Process Computer was given in Section 1.2.1.1.

In addition, the Post Trip data can be obtained in the Control Room on two of three terminets, or in computer room on the line printer or on either teriminet.

Historical data may be obtained on either the color video or the terminet. Also, historical data can be obtained from the list of dedicated strip charts which are located in the control room (see Section 1.2.1.1).

1.2.2.2 Parameters monitored, sampling rate and basis for selecting parameters and sampling rate.

NMP2 Response

The parameters monitored by the Process Monitoring System (PMS) are located on Attachments 4 "NSSS Post Trip Log" and 5 "BOP Post Trip log".

These NSSS points provide information to enable the system to calculate and display or printout, a variety of nuclear system data arrays (LPRM readings, sensitivities, and calibration constants; APRM gain adjustment factors and trip levels; control rod positions; fuel bundle isotopic compositions, etc.).

The BOP points provide data to enable the system to perform calculations, evaluations of the status and efficiency of various plant systems not directly related to the nuclear steam supply. The calculations include turbine cycle performance, condenser performance, unit electric performance, and feedwater heater performance.

Selected Nuclear Steam Supply System and Balance of Plant digital signals are scanned once each second for the purposes of monitoring process variable alarms. Each time an input is scanned, it is compared to its previous state and if it is different, the program will determine the nature of the change, (e.g., alarm or return-to-normal) and a descriptive message will be logged.

1.2.2.3 Duration of time history (minutes before trip and minutes after trip).

NMP2 Response

Two Post-Trip Logs are used at Nine Mile Unit 2. The first log is an accumulation of points associated with the Nuclear Steam Supply System, and the second is an accumulation of Balance of Plant points. This log's data interval is from 5 minutes before the trip until 5 minutes after. The BOP log is made up of a maximum of 48 predetermined points. The data collection period ranges from 30 minutes before the trip until 30 minutes after the event. Each point is scanned every 15 seconds. Both logs will be initiated upon completion of their recording constraints. Recording and scan rate times are changeable via software change routines to allow plant operations to vary the process monitoring function.

1.2.2.4 Format for displaying data including scale (readability) of time histories.

NMP2 Response

Attachment 6 is a representation of what both the NSSS and BOP Post-Trip logs look like. This attachment is self-explanatory as to the data that is contained on these logs.

1.2.2.5 Capability for retention of data, information and physical evidence (both hardware and software).

NMP2 Response

Post-Trip logs can be recovered in the same manner as discussed in Section 1.2.1.5. The only difference being that the logs can be recovered and reprinted exactly as the original log. Post-Trip logs can be demanded at a later time if no other event has generated another new Post-Trip log to overlay existing data..

1.2.2.6 Power source(s) (e.g., Class 1E, non-Class 1E, noninterruptible).

NMP2 Response

Power sources are the same sources discussed in Section 1.2.1.6.

1.2.3 Other data and information provided to assess the cause of unscheduled reactor shutdowns.

NMP2 Response

Other data and information available to assess the cause of unscheduled reactor shutdowns include operator logs, trend recorders, meter indications, surveillance test data sheets, seismic recording equipment, operator interviews, and occurrence reports. Other computer systems available to assist in the evaluation of

1.2.3 NMP2 Response (Cont'd)

unscheduled shutdowns are the Safety Parameter Display. System (SPDS), and the General Electric's Transient Analysis Recording System (GETAR's). In addition, previous scram report data and information is available at the operator's disposal enabling them to compare event information with known or expected plant behavior.

1.2.4 Schedule for any planned changes to existing data and information capability.

NMP2 Response

The SOE printout will be changed from 30 to 5 seconds to be more consistent with timelines of plant data for operator response. Also, points will be added to SOE & Alarm displays as required for more effective plant operations.

ATTACHMENT A

Main Control Room Pen Recorders (strip charts)

- Reactor Vessel Level-Fuel Zone
- Post Accident Monitor, channels (A&B) Rx level, Rx Pressure
- Reactor Water Cleanup F/D Inc. Conductivity & Oxygen sample
- Service Water/RHR Temperature
- RECIRC Pumps Suction Temperature
- Total RECIRC Flow
- Reactor Pressure; Turbine Steam Flow
- Core Pressure Drop; Total Flow
- Reactor Steam Flow; Feedwater Flow
- Reactor Water Level
- Condensate Demineralized Conductivity in/out & oxygen out
- Inlet Conductivity High; out Conductivity High
- 6th Point Heater Outlet Conductivity & Oxygen/PG
- Generator Turbine Component Position
- Core Monitoring
- Bearing Metal Temperatures
- Turbine Temperatures
- Turbine Vibration
- Bearing Drain & Thrust Bearing Temperatures
- CRD Pump Discharge Conductivity and Oxygen
- IRM/APRM Recorders (4 units)
- SRM channel (records two of 4 channels)
- Main Steam Reheater Reheating Steam Supply Temperature 1A & 1B
- Circulating Water System Return Water Conductivity & PH
- Main Generator Frequency
- 345KV Line Main Generator Volts

ATTACHMENT 1

03/08/85

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM 21IIC- 6
CONTROLLED ISSUE - REVISION 9

R V	PT ID / ESK	LSK	POINT DESCRIPTION / VENDOR REF DWG	P N / T T	PR-0 / PR-1	A S / S E	CO USED / ALM NO	CUT-OUT 1 POINT ID / ANN WIND	COI IS / VOLT	SI IS	SOURCE / DESTINATION	TROUBLE CONTACT
	10 ISSUC15		TB STOP V FAST CLOS CH B		ALMCLR	1	0			1	H13-P630-B	3A-0431
		N/A	7.225-001-019	0 0	TRIP	1	0431	CEC603406	1		C91-P623-4-12-3,4	
	10 ISSUC16		TB STOP V FAST CLOS CH D		ALMCLR	1	0			1	H13-P630-B	3A-0432
		N/A	7.225-001-019	0 0	TRIP	1	0432	CEC603406	1		C91-P623-4-12-5,6	
	10 NMEUC01		NEUT MON SYS CH A TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0437
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0437	CEC603102	1		C91-P623-4-12-7,8	
	15 NMEUC02		NEUT MON SYS CH C TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0438
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0438	CEC603102	1		C91-P623-4-12-9,10	
	15 NMEUC03		NEUT MON SYS CH B TRIP		ALMCLR	1	0			1	H13-P630-B	3A-0439
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0439	CEC603402	1		C91-P623-4-12-11,12	
	10 NMEUC04		NEUT MON SYS CH D TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0440
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0440	CEC603402	1		C91-P623-4-12-13,14	
	13 NMIUC01		IRM CHAN A UPSC LVL TRIP		ALMCLR	1	0			1	H13-P630-B	3A-0298
	N/A	N/A	7.224-001-028	0 0	TRIP	1	0298	CEC603201	1		C91-P623-4-01-1,2	
	12 NMIUC02		IRM CHAN B UPSC LVL TRIP		ALMCLR	1	0			1	H13-P630-B	3A-0302
	N/A	N/A	7.224-001-028	0 0	TRIP	1	0302	CEC603301	1		C91-P623-4-01-3,4	
	07 NMIUC03		IRM CHAN C UPSC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0711
	N/A	N/A	7.224-001-028	0 0	TRIP	1	0711	CEC603201	1		C91-P623-4-01-5,6	
	07 NMIUC04		IRM CHAN D UPSC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0712
	N/A	N/A	7.224-001-028	0 0	TRIP	1	0712	CEC603301	1		C91-P623-4-01-7,8	
	07 NMIUC05		IRM CHAN E UPSC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0713
	N/A	N/A	7.224-001-028	0 0	TRIP	1	0713	CEC603201	1		C91-P623-4-01-9,10	
	07 NMIUC06		IRM CHAN F UPSC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0714
	N/A	N/A	7.224-001-028	0 0	TRIP	1	0714	CEC603301	1		C91-P623-4-01-11,12	
	07 NMIUC07		IRM CHAN G UPSC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0715
	N/A	N/A	7.224-001-028	0 0	TRIP	1	0715	CEC603201	1		C91-P623-4-01-13,14	
	07 NMIUC08		IRM CHAN H UPSC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0716
	N/A	N/A	7.224-001-028	0 0	TRIP	1	0716	CEC603301	1		C91-P623-4-01-15,16	
	10 NMPUC01		APRM CH A USPC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0307
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0307	CEC603202	1		C91-P623-4-03-1,2	
	10 NMPUC02		APRM CH B USPC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0308
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0308	CEC603302	1		C91-P623-4-03-3,4	
	10 NMPUC03		APRM CH C USPC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0717
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0717	CEC603202	1		C91-P623-4-03-5,6	
	10 NMPUC04		APRM CH D USPC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0718
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0718	CEC603302	1		C91-P623-4-03-7,8	
	10 NMPUC05		APRM CH E USPC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0719
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0719	CEC603202	1		C91-P623-4-03-9,10	
	10 NMPUC06		APRM CH F USPC LVL TRIP		ALMCLR	1	0			1	H13-P630-D	3A-0720
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0720	CEC603302	1		C91-P623-4-03-11,12	

ATTACHMENT 1B

03/08/85

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM ZIMC- 6
CONTROLLED ISSUE - REVISION 9

R V	PT ID / ESN	POINT DESCRIPTION / LSN	VENDOR REF DWG	P T / H T	PR-0 / PR-1	A S / S E	CO USED / ALM NO	CUT-OUT 1 POINT ID / ANN WIND	CUT IS / VOLT	SOURCE / DESTINATION	TROUBLE CONTACT
	10 NMPUC07	APRM CH A USPC THRM STAT			NORMAL		1	0		1 H13-P630-B	3A-0756
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0756	CEC603202	1	C91-P623-4-14-1,2	
	10 NMPUC08	APRM CH B USPC THRM STAT			NORMAL		1	0		1 H13-P630-B	3A-0757
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0757	CEC603302	1	C91-P623-4-14-3,4	
	10 NMPUC09	APRM CH C USPC THRM STAT			NORMAL		1	0		1 H13-P630-B	3A-0758
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0758	CEC603202	1	C91-P623-4-14-5,6	
	10 NMPUC10	APRM CH D USPC THRM STAT			NORMAL		1	0		1 H13-P630-B	3A-0759
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0759	CEC603302	1	C91-P623-4-14-7,8	
	10 NMPUC11	APRM CH E USPC THRM STAT			NORMAL		1	0		1 H13-P630-D	3A-0760
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0760	CEC603202	1	C91-P623-4-14-9,10	
	10 NMPUC12	APRM CH F USPC THRM STAT			NORMAL		1	0		1 H13-P630-D	3A-0761
	N/A	N/A	7.224-001-078	0 0	TRIP	1	0761	CEC603302	1	C91-P623-4-14-11,12	
	12 RDSUC05	DISCH VOL H HTR LVL CH A			ALMCLR		1	0		1 H13-P630-C	3A-0397
	N/A	N/A	7.225-001-014	0 0	TRIP	1	0397	CEC603109	1	C91-P623-4-05-1,2	
	12 RDSUC06	DISCH VOL H HTR LVL CH C			ALMCLR		1	0		1 H13-P630 C	3A-0398
	N/A	N/A	7.225-001-014	0 0	TRIP	1	0398	CEC603109	1	C91 P623-4-05-3,4	
	12 RDSUC07	DISCH VOL H HTR LVL CH B			ALMCLR		1	0		1 H13-P630 C	3A-0399
	N/A	N/A	7.225-001-014	0 0	TRIP	1	0399	CEC603409	1	C91 P623 4 05-5,6	
	12 RDSUC08	DISCH VOL H HTR LVL CH D			ALMCLR		1	0		1 H13 P630 D	3A-0400
	N/A	N/A	7.225-001-014	0 0	TRIP	1	0400	CEC603409	1	C91 P623 4 05 7,8	
	12 RHSDC20	RHR PUMP 1A STATUS			STOP		1	0		1 H13 P630 B	52-2RHSA01
	5RHS04	N/A	7.241-001-021	0 0	RUN	1			1	C91 P623 4 07 1,2	
	12 RHSDC21	RHR PUMP 1B STATUS			STOP		1	0		1 H13 P630-B	52-2RHSB01
	5RHS05	N/A	7.241-001-021	0 0	RUN	1			1	C91-P623-4-07-3,4	
	12 RHSDC22	RHR PUMP 1C STATUS			STOP		1	0		1 H13-P630-D	52-2RHSC01
	5RHS03	N/A	7.241-001-021	0 0	RUN	1			1	C91-P623-4-07-5,6	
	12 RHSDC23	RHR LOOP A PMP DISH PRES			NORMAL		1	0		1 H13-P630-D	B22C-N14A
	N/A	N/A	7.241-001-032	0 0	HIGH	1			1	C91-P623-4-07-7,8	
	13 RHSBC24	RHR LOOP B PMP DISH PRES			NORMAL		1	0		1 H13-P630-D	B22C-N13B
	N/A	N/A	7.212-001-032	0 0	HIGH	1			1	C91-P623-4-07-9,10	
	13 RHSBC25	RHR LOOP C PMP DISH PRES			NORMAL		1	0		1 H13-P630-B	B22C-N14B
	N/A	N/A	7.212-001-032	0 0	HIGH	1			1	C91-P623-4-07-11,12	
	12 RHSBC26	RHR INJECT FLOW LOOP A			OFF		1	0		1 H13-P630-D	E12-N112A
	N/A	N/A	7.241-001-021	0 0	ON	1			1	C91-P623-4-07-13,14	
	12 RHSBC27	RHR INJECT FLOW LOOP B			OFF		1	0		1 H13-P630-B	E12-N112B
	N/A	N/A	7.241-001-021	0 0	ON	1			1	C91-P623-4-07-15,16	
	12 RHSBC28	RHR INJECT FLOW LOOP C			OFF		1	0		1 H13-P630-B	E12-N112C
	N/A	N/A	7.241-001-021	0 0	ON	1			1	C91-P623-4-09-1,2	
	15 RPSUC01	MANUAL SCRAM DIV 1			ALMCLR		1	0		1 H13-P630-D	3A-0445
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0445	CEC603111	1	C91-P623-4-05-9,10	

ATT[illegible]ENT 1C

03/08/85

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM 2MC- 6
CONTROLLED ISSUE - REVISION 9

R V	PT ID		POINT DESCRIPTION		PR-O	A S	CO USED	CUT-OUT 1 POINT ID	CUT IS	SI TS	SOURCE	TROUBLE CONTACT
	ESN	LSN	VENDOR REF DWG	P T N T	PR-1	S E	ALM NO	ANN WIND		VO LT	DESTINATION	
15	RPSUC02		MANUAL SCRAM DIV 2		ALMCLR		1	0		1	H13-P630-D	3A-0447
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0447	CEC603411		1	C91-P623-4 05-11,12	
15	RPSUC03		REACTOR SCRAM DIV 1		ALMCLR		1	0		1	H13-P630-D	3A-0441
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0441	CEC603110		1	C91-P623-4-05-13,14	
15	RPSUC04		REACTOR SCRAM DIV 2		ALMCLR		1.	0		1	H13 P630 D	3A-0442
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0442	CEC603410		1	C91 P623 4 05 15,16	
15	RPSUC05		MANUAL SCRAM DIV 3		ALMCLR		1	0		1	H13 P630 D	3A 0446
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0446	CEC603111		1	C91 P623-4 13 7,8	
15	RPSUC06		MANUAL SCRAM DIV 4		ALMCLR		1	0		1	H13 P630 D	3A-0448
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0448	CEC603411		1	C91-P623-4-13 9,10	
15	RPSUC07		REACTOR SCRAM DIV 3		ALMCLR		1	0		1	H13-P630-D	3A-0443
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0443	CEC603110		1	C91-P623-4-13-11,12	
15	RPSUC08		REACTOR SCRAM DIV 4		ALMCLR		1	0		1	H13-P630-B	3A-0449
	N/A	N/A	7.225-001-015	0 0	TRIP	1	0449	CEC603410		1	C91-P623-4-13-13,14	
13	RRSBC09		RRCS CONFIRMED ATWS DIV1		ALMCLR		1			1	H13-P630-D	3A-0850
	N/A	N/A	7.226-001-029	0 0	ALARM	1	0850	CEC603439		1	C91-P623-4-15-5,6	
13	RRSBC10		RRCS CONFIRMED ATWS DIV2		ALMCLR		1			1	H13-P630-D	3A-0865
	N/A	N/A	7.226-001-031	0 0	ALARM	1	0865	CEC603439		1	C91-P623-4-15-7,8	
13	RRSBC13		RRCS MAN INITATED DIV 1		ALMCLR		1			1	H13-P630-D	3A-0854
	N/A	N/A	7.226-001-030	0 0	ALARM	1	0854	CEC603432		1	C91-P623-4-15-1,2	
13	RRSBC14		RRCS MAN INITATED DIV 2		ALMCLR		1			1	H13-P630-D	3A-0869
	N/A	N/A	7.226-001-032	0 0	ALARM	1	0869	CEC603432		1	C91-P623-4-15 3,4	
15	SVVBC01		RLF VLV PILOT SOL NO A		CLOSED		1			1	H13-P630-C	
	N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-09-3,4	
15	SVVBC02		RLF VLV PILOT SOL NO B		CLOSED		1			1	H13-P630-C	
	N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-09-5,6	
15	SVVBC03		RLF VLV PILOT SOL NO C		CLOSED		1			1	H13-P630-C	
	N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-09-7,8	
15	SVVBC04		RLF VLV PILOT SOL NO D		CLOSED		1			1	H13 P630-C	
	N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623 4-09-9,10	
15	SVVBC05		RLF VLV PILOT SOL NO E		CLOSED		1			1	H13 P630 C	
	N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-09-11,12	
15	SVVBC06		RLF VLV PILOT SOL NO F		CLOSED		1			1	H13-P630 C	
	N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-09-13,14	
15	SVVBC07		RLF VLV PILOT SOL NO G		CLOSED		1			1	H13-P630-C	
	N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623 4-07 15,16	
15	SVVBC08		RLF VLV PILOT SOL NO H		CLOSED		1			1	H13-P630-C	
	N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623 4-11 1,2	
15	SVVBC09		RLF VLV PILOT SOL NO J		CLOSED		1			1	H13-P630-C	
	N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-11-3,4	

ATTACHMENT 1D

03/08/85

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM 23HC- 4
CONTROLLED ISSUE - REVISION 9

R V / PT ID	POINT DESCRIPTION			PR-0	A S	CO USED	CUT-OUT 1 POINT ID	CUT IS	ST IS	SOURCE	INHIBIT CONTACT
ESK	LSK	VENDOR REF DWG	P H T T	PR-1	S E	ALM NO	ANN WIND		VO LT	DESTINATION	
15 SVVBC10	RLF VLV PILOT SOL NO K			CLOSED		1			1	H13-P630-C	
N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-11-5,6	
15 SVVBC11	RLF VLV PILOT SOL NO L			CLOSED		1			1	H13-P630-C	
N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-11-7,8	
15 SVVBC12	RLF VLV PILOT SOL NO M			CLOSED		1			1	H13-P630-C	
N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-11-9,10	
15 SVVBC13	RLF VLV PILOT SOL NO N			CLOSED		1			1	H13-P630-C	
N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-11-11,12	
15 SVVBC14	RLF VLV PILOT SOL NO P			CLOSED		1			1	H13-P630-C	
N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-11-13,14	
15 SVVBC15	RLF VLV PILOT SOL NO R			CLOSED		1			1	H13-P630-C	
N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-11-15,16	
15 SVVBC16	RLF VLV PILOT SOL NO S			CLOSED		1			1	H13-P630-C	
N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-13-1,2	
15 SVVBC17	RLF VLV PILOT SOL NO U			CLOSED		1			1	H13-P630-C	
N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4-13-3,4	
15 SVVBC18	RLF VLV PILOT SOL NO V			CLOSED		1			1	H13-P630-C	
N/A	N/A	7.412-001-032	0 0	OPEN	1				1	C91-P623-4 13 5,6	
14 TMAUC03	BACKUP OVERSPEED TRIP			ALMCLR		1	0		1	H13-P850-A	3A-3981
10IHA439	1-4B	7.330-002-019	0 0	TRIP	1	3981	CLC051109		1	C91-P623-3-00-1,2	
15 TMAUC05	LOSS OF STAT COOL TRIP			RESET		1	0		1	H13-P850-A	TDF13-819
10IHA439	N/A	7.330-002-017	0 0	TRIP	1				1	C91-P623-3-00-3,4	
14 TMAUC07	VACUUM TRIP			ALMCLR		1	0		1	H13-P850-A	3A-3985
10IHA439	1-4C	7.330-002-018	0 0	TRIP	1	3985	CLC051109		1	C91-P623-3-00-5,6	
15 TMAUC09	OVERSPEED TRIP			RESET		1	0		1	H13-P850-A	TDF17-819
10IHA440	N/A	7.330-002-020	0 0	TRIP	1				1	C91-P623-3-00-7,8	
14 TMAUC11	CUST TRIP			ALMCLR		1	0		1	H13-P850 A	3A-3989
10IHA440	1-4D	7.330-002-017	0 0	TRIP	1	3989	CLC851109		1	C91-P623-3-00-9,10	
14 TMAUC12	TSI VIBRATION HIGH TRIP			ALMCLR		1	0		1	H13-P850 A	3A-4001
10IHA440	1-4A	7.330-002-017	0 0	TRIP	1	4001	CLC651119		1	C91-P623-3-00 11,12	
14 TMAUC14	TURB EMERG MANUAL TRIP			ALMCLR		1	0		1	H13-P850 A	TDF21-9810
10IHA440	1-4C	7.330-002-019	0 0	TRIP	1				1	C91-P623-3-00-13,14	
15 TMAUC31	MASTER TRIP TRIPPED			ALMCLR		1	0		1	H13 P850 A	TDF17-1,2
10IHA439		7.330-002-019	0 0	TRIP	1				1	C91-P623-3-00 15,16	

ATTACHMENT 1E

03/08/85

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM 2IHC- 6
CONTROLLED ISSUE - REVISION 9

R V / PT ID / ESK	POINT DESCRIPTION / LSK	VENDOR REF DWG	P H	T T	PR-0 / PR-1	A S / S E	CO USED / ALM NO	CUT-OUT 1 POINT ID / ALM MIND	COL IS / SI IS / VO LT	SOURCE / DESTINATION	TROUBLE CONTACT
11 CSHLC10	HPCS LO RX H2O LVL CH A				NORMAL	1	0			1 H13-P630-C	3A-0100
N/A	N/A	7.243-001-012	0	0	LOW	1	0180	CEC601708	1	C91-P623-4-04-5,6	
11 CSHLC11	HPCS LO RX H2O LVL CH B				NORMAL	1	0			1 H13-P630-C	3A-0101
N/A	N/A	7.243-001-012	0	0	LOW	1	0181	CEC601708	1	C91-P623-4-04-7,8	
11 CSHLC12	HPCS LO RX H2O LVL CH C				NORMAL	1	0			1 H13-P630-C	3A-0102
N/A	N/A	7.243-001-012	0	0	LOW	1	0182	CEC601708	1	C91-P623-4-04-9,10	
11 CSHLC13	HPCS LO RX H2O LVL CH D				NORMAL	1	0			1 H13-P630-C	3A-0103
N/A	N/A	7.243-001-012	0	0	LOW	1	0183	CEC601708	1	C91-P623-4-04-11,12	
10 CSHPC06	HPCS DW PRESS CHAN A				NORMAL	1	0			1 H13-P630-C	3A-0175
N/A	N/A	7.243-001-012	0	0	HIGH	1	0175	CEC601707	1	C91-P623-4-02-13,14	
10 CSHPC07	HPCS DW PRESS CHAN B				NORMAL	1	0			1 H13-P630-C	3A-0176
N/A	N/A	7.243-001-012	0	0	HIGH	1	0176	CEC601707	1	C91-P623-4-[illegible]-15,16	
10 CSHPC08	HPCS DW PRESS CHAN C				NORMAL	1	0			1 H13-P630-C	3A-0177
N/A	N/A	7.243-001-012	0	0	HIGH	1	0177	CEC601707	1	C91-P623-4-04-1,2	
10 CSHPC09	HPCS DW PRESS CHAN D				NORMAL	1	0			1 H13-P630-C	3A-0178
N/A	N/A	7.243-001-012	0	0	HIGH	1	0178	CEC601707	1	C91-P623-4-04-3,4	
15 CSLBC10	LPCS PUMP BREAKER				OPEN	1	0			1 H13-P630-B	52-2CSLN01
5CSL01	N/A	7.242-001-004	0	0	CLOSED	1			1	C91-P623-4-00-7,NC	
14 CSLBC11	LPCS SYSTEM PRESSURE				NORMAL	1	0			1 H13-P630-B	B22-N13A
N/A	N/A	7.242-001-004	0	0	HIGH	1			1	C91-P623-4-00-9,10	
15 CSLBC12	LPCS INJECTION FLOW				NORMAL	1	0			1 H13-P630-B	E21A-N51
N/A	N/A	7.242-001-004	0	0	LOW	1			1	C91-P623-4-00-11,NC	
10 CSLLC01	LPCS/RHR/ADS HTR LVL C-A				NORMAL	1	0			1 H13-P630-C	3A-0070
N/A	N/A	7.242-001-004	0	0	LOW	1	0070	CEC601403	1	C91-P623-4-00-13,14	
11 CSLLC02	LPCS/RHR/ADS HTR LVL C-E				NORMAL	1	0			1 H13-P630-C	3A-0071
N/A	N/A	7.242-001-004	0	0	LOW	1	0071	CEC601403	1	C91-P623-4-00-15,16	
10 CSLPC07	LPCS/RHR/ADS DW PR CH A				NORMAL	1	0			1 H13-P630-C	3A-0068
N/A	N/A	7.242-001-004	0	0	HIGH	1	0068	CEC601402	1	C91-P623-4-02-5,6	
11 CSLPC08	LPCS/RHR/ADS DW PR CH E				NORMAL	1	0			1 H13-P630-C	3A-0069
N/A	N/A	7.242-001-004	0	0	HIGH	1	0069	CEC601402	1	C91-P623-4-02-7,8	
15 EGPIC15	SPARE									1 H13-P853-B	
101HA408						1		CEC052502	1	C91-P626-2-03-1,2	
15 EGPIC16	SPARE									1 H13-P853-B	
101HA903						1		CEC852502	1	C91-P626-4-05-3,4	
15 EHSBC25	LOAD SHED SIGNAL BUS 101				ALMCLR	1	0			1 H13-P[illegible]	3A-EHSBC25
101HA103A	29-9.4A		0	0	ALARM	1			2	C91-P873-3-12-1,2	
15 EHSBC26	LOAD SEQ SIG BUS 101				ALMCLR	1	0			1 H13-P[illegible]	3A-EHSBC26
101HA103A	29-9.4B		0	0	ALARM	1			2	C91-P873-3-12-3,4	
15 EHSBC27	LOAD SEQ SIG BUS 101				ALMCLR	1	0			1 H13-P[illegible]	3A-EHSBC27
101HA103B	29-9.4B		0	0	ALARM	1			2	C91-P623-3-12-5,6	

ATTACHMENT 1F

03/08/85

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM 2IHC- 6
CONTROLLED ISSUE - REVISION 9

R V / ESK	PT ID / LSK	POINT DESCRIPTION / VENDOR REF DWG	PT HT	PR-0 / PR-1	AS / SE	CO USED / ALM NO	CUT-OUT 1 POINT ID / ANN WIND	CO1 IS	SI IS / VO LT	SOURCE / DESTINATION	TROUBLE CONTACT
15 ENSBC29		G1 ACB101-1 TR SIGNAL		ALMCLR		1	0			1 H13 P0500	3A-ENSBC29
101HA103A	24-9.4A		0 0	ALARM	1				2	C91 P623 3 12 7,8	
15 ENSBC30		DIV 1 LOCA SIGNAL		ALMCLR		1	0			1 H13 P0580	3A-ENSBC30
101HA103A	24-9.4D		0 0	ALARM	1				2	C91-P623-3-12-9,10	
15 ENSBC32		D1 BLK IRL CLOS SFPC-SHP		AVAIL		1	0			1 H13 P0520	3A-ENSBC32
101HA104A	24.9-4B		0 0	NOAVLB	1				2	C91-P623-3-12-11,12	
15 ENSBC35		LO SHED SIGNAL BUS 103		ALMCLR		1	0			1 H13-P0500	3A-ENSBC35
101HA201B	24-9.4A		0 0	ALARM	1				2	C91-P623-3-12-13,14	
15 ENSBC36		LO SEQ SIG LOCA BUS 103		ALMCLR		1	0			1 H13-P0580	3A-ENSBC36
101HA201B	24-9.4D		0 0	ALARM	1				2	C91-P623-3-12-15,16	
15 ENSBC37		LO SEQ SIG BUS 103		ALMCLR		1	0			1 H13-P8530	3A-ENSBC37
101HA203A	24-9.4B		0 0	ALARM	1				2	C91-P623-3-13-1,2	
15 ENSBC39		G3 ACB103-1 TR SIGNAL		ALMCLR		1	0			1 H13-P8550	3A-ENSBC39
101HA203A	24-9.4A		0 0	ALARM	1				2	C91-P623-3-13-3,4	
15 ENSBC40		DIV 2 LOCA SIGNAL		ALMCLR		1	0			1 H13-P0560	3A-ENSBC40
101HA203A	24-9.4D		0 0	ALARM	1				2	C91-P623-3-13-5,6	
15 ENSBC42		D2 BLK IRL CLOS SFPC-SHP		AVAIL		1	0			1 H13-P0520	3A-ENSBC42
101HA203A	24-9.4B		0 0	NOAVLB	1				2	C91-P623-3-13-7,8	
11 ISCBC03		RHR/ADS HTR LVL CHAN B		NORMAL		1	0			1 H13-P630-C	3A-0115
N/A	N/A	7.241-001-020	0 0	LOW	1	0115	CEC601406		1	C91-P623-4-02-1,2	
11 ISCBC04		RHR/ADS WATER LEVELCH F		NORMAL		1	0			1 H13-P630-C	3A-0119
N/A	N/A	7.241-001-020	0 0	LOW	1	0114	CEC601406		1	C91-P623-4-02-3,4	
05 ISCBC07		RHR/ADS DW PRESS CHAN B		NORMAL		1	0			1 H13 P630-C	3A-0113
N/A	N/A	7.241-001-020	0 0	HIGH	1	0113	CEC601405		1	C91-P623-4-02-9,10	
11 ISCBC08		RHR/ADS DW PRESS CH F		NORMAL		1	0			1 H13-P630-C	3A-0112
N/A	N/A	7.241-001-020	0 0	HIGH	1	0112	CEC601405		1	C91-P623-4-02-11,12	
14 ISCUC01		RX LO WATER LEVEL CHAN A		ALMCLR		1	0			1 H13-P630-C	3A-0413
N/A	N/A	7.225-001-014	0 0	TRIP	1	0413	CEC603105		1	C91-P623-4-04-13,14	
14 ISCUC02		RX LO WATER LEVEL CHAN C		ALMCLR		1	0			1 H13-P630-C	3A-0414
N/A	N/A	7.225-001-014	0 0	TRIP	1	0414	CEC603105		1	C91-P623-4-04-15,16	
14 ISCUC03		REAC LOW WTR LVL CHAN B		ALMCLR		1	0			1 H13-P630-C	3A-0415
N/A	N/A	7.225-001-014	0 0	TRIP	1	0415	CEC603405		1	C91-P623-4-05-1,2	
14 ISCUC04		REAC LOW WTR LVL CHAN D		ALMCLR		1	0			1 H13-P630-C	3A-0416
N/A	N/A	7.225-001-014	0 0	TRIP	1	0416	CEC603405		1	C91-P623-4-06-3,4	
14 ISCUC05		REAC CHAN A HIGH PRSS TR		ALMCLR		1	0			1 H13-P630-C	3A-0409
N/A		7.225-001-014	0 0	TRIP	1	0409	CEC603103		1	C91-P623-4-05-5,6	
14 ISCUC06		REAC CHAN C HI PR TRIP		ALMCLR		1	0			1 H13 P630-C	3A-0410
N/A		7.225-001-014	0 0	TRIP	1	0410	CEC603103		1	C91 P623-4-05-7,8	
14 ISCUC07		REAC CHAN D HI PR TRIP		ALMCLR		1	0			1 H13 P630-C	3A-0411
N/A		7.225-001-014	0 0	TRIP	1	0411	CEC603403		1	C91-P623-4-06-9,10	

ATTACHMENT 16

03/08/85

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM 2MC- 6
CONTROLLED ISSUE - REVISION 9

R V	PT ID / ESK	LSK	POINT DESCRIPTION / VENDOR REF DWG	P T H T	PR-0 / PR-1	A S / S E	CO USED / ALM NO	CUT-OUT 1 POINT ID / ANN WIND	COI IS / VO LT	SI TS	SOURCE / DESTINATION	TROUBLE CONTACT
14	ISCUC08		REAC CHAN D HIGH PRSS TR		ALMCLR		1	0		1	H13-P630-C	3A-0412
	N/A		7.225-001-014	0 0	TRIP	1	0412	CEC603403	1		C91-P623-4-08-11,12	
14	ISCUC09		DRYWELL HI PR CHAN A		ALMCLR		1	0		1	H13-P630-C	3A-0405
	N/A		7.225-001-014	0 0	TRIP	1	0405	CEC603101	1		C91-P623-4-08-13,14	
14	ISCUC10		DRYWELL HI PR CHAN C		ALMCLR		1	0		1	H13-P630-C	3A-0406
	N/A		7.225-001-014	0 0	TRIP	1	0406	CEC603101	1		C91-P623-4-08-15,16	
14	ISCUC11		DRYWELL HI PR CHAN B		ALMCLR		1	0		1	H13-P630-C	3A-0407
	N/A		7.225-001-014	0 0	TRIP	1	0407	CEC603401	1		C91-P623-4-08-1,2	
14	ISCUC12		DRYWELL HI PR CHAN D		ALMCLR		1	0		1	H13-P630-C	3A-0408
	N/A		7.225-001-014	0 0	TRIP	1	0408	CEC603401	1		C91-P623-4-08-3,4	
15	HSSBC20		TURBINE BYPASS VALVE		CLOSED		0	0		1	H13-P630-C	
	N/A	3-1J	1.010-002-072	0 0	OPEN	1			1		C91-P623-4-08-5,6	
10	HSSUC01		MSL ISOL V CLOS CHAN A		ALMCLR		1	0		1	H13-P630-C	3A-0401
		N/A	7.225-001-014	0 0	TRIP	1	0401	CEC603108	1		C91-P623-4-08-7,8	
15	HSSUC02		MSL ISOL V CLOS CHAN C		ALMCLR		1	0		1	H13-P630-C	3A-0403
		N/A	7.225-001-014	0 0	TRIP	1	0403	CEC603108	1		C91-P623-4-08-9,10	
15	HSSUC03		MSL ISOL V CLOS CHAN B		ALMCLR		1	0		1	H13-P630-C	3A-0402
		N/A	7.225-001-014	0 0	TRIP	1	0402	CEC603408	1		C91-P623-4-08-11,12	
10	HSSUC04		MSL ISOL V CLOS CHAN D		ALMCLR		1	0		1	H13-P630-C	3A-0404
		N/A	7.225-001-014	0 0	TRIP	1	0404	CEC603408	1		C91-P623-4-08-13,14	
10	HSSUC05		MN STM LN CHAN A RADN H		ALMCLR		1	0		1	H13-P630-B	3A-0417
		N/A	7.225-001-014	0 0	TRIP	1	0417	CEC603107	1		C91-P623-4-08-15,16	
10	HSSUC06		MN STM LN CHAN C RADN H		ALMCLR		1	0		1	H13-P630-B	3A-0418
		N/A	7.225-001-014	0 0	TRIP	1	0418	CEC603107	1		C91-P623-4-10-1,2	
10	HSSUC07		MN STM LN CHAN B RADN H		ALMCLR		1	0		1	H13-P630-B	3A-0419
		N/A	7.225-001-014	0 0	TRIP	1	0419	CEC603407	1		C91-P623-4-10-3,4	
10	HSSUC08		MN STM LN CHAN D RADN H		ALMCLR		1	0		1	H13-P630-B	3A-0420
		N/A	7.225-001-014	0 0	TRIP	1	0420	CEC603407	1		C91-P623-4-10-5,6	
12	HSSUC09		TB CONT V FAST CLOS CH A		ALMCLR		1	0		1	H13-P630-B	3A-0421
		N/A	7.225-001-014	0 0	TRIP	1	0421	CEC603104	1		C91-P623-4-10-7,8	
12	HSSUC10		TB CONT V FAST CLOS CH C		ALMCLR		1	0		1	H13-P630-B	3A-0422
		N/A	7.225-001-014	0 0	TRIP	1	0422	CEC603104	1		C91-P623-4-10-9,10	
10	HSSUC11		TB CONT V FAST CLOS CH B		ALMCLR		1	0		1	H13-P630-B	3A-0423
		N/A	7.225-001-014	0 0	TRIP	1	0423	CEC603404	1		C91-P623-4-10-11,12	
10	HSSUC12		TB CONT V FAST CLOS CH D		ALMCLR		1	0		1	H13-P630-B	3A-0424
		N/A	7.225-001-014	0 0	TRIP	1	0424	CEC603404	1		C91-P623-4-10-13,14	
10	HSSUC13		TB STOP V FAST CLOS CH A		ALMCLR		1	0		1	H13-P630-B	3A-0429
		N/A	7.225-001-014	0 0	TRIP	1	0429	CEC603106	1		C91-P623-4-10-15,16	
10	HSSUC14		TB STOP V FAST CLOS CH C		ALMCLR		1	0		1	H13-P630-B	3A-0430
		N/A	7.225-001-014	0 0	TRIP	1	0430	CEC603106	1		C91-P623-4-12-1,2	

ATTACHMENT 1H

03/08/85

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM 21NC- 6
CONTROLLED ISSUE - REVISION 9

R V	PT ID / ESK	LSK	POINT DESCRIPTION / VENDOR REF DWG	P T / N T	PR-0 / PR-1	A S / S E	CO USED / ALM NO	CUT-OUT 1 POINT ID / ANN WIND	COI IS	SI TS / VO LT	SOURCE / DESTINATION	TROUBLE CONTACT
07	AAAXC71		SPARE - SOE							1		
						1				2	C91-P623-3-11-9.10	
07	AAAXC72		SPARE - SOE							1		
						1				2	C91-P623-3-11-11.12	
07	AAAXC73		SPARE - SOE							1		
						1				2	C91-P623-3-11-13.14	
07	AAAXC74		SPARE - SOE							1		
						1				2	C91-P623-3-11-15.16	
07	AAAXC87		SPARE - SOE							1		
						1				2	C91-P623-3-13-9.10	
07	AAAXC88		SPARE - SOE							1		
						1				2	C91-P623-3-13-11.12	
07	AAAXC89		SPARE - SOE							1		
						1				2	C91-P623-3-13-13.14	
07	AAAXC90		SPARE - SOE							1		
						1				2	C91-P623-3-13-15.16	
07	AAAXC91		SPARE - SOE							1		
						1				2	C91-P623-3-14-1.2	
07	AAAXC92		SPARE - SOE							1		
						1				2	C91-P623-3-14 3.4	
07	AAAXC93		SPARE - SOE							1		
						1				2	C91-P623-3-14 5.6	
07	AAAXC94		SPARE - SOE							1		
						1				2	C91 P623 3-14 7.8	
07	AAAXC95		SPARE - SOE							1		
						1				2	C91 P623-3-14-9.10	
07	AAAXC96		SPARE - SOE							1		
						1				2	C91-P623-3-14 11.12	
07	AAAXC97		SPARE - SOE							1		
						1				2	C91-P623-3-14-13.14	
07	AAAXC98		SPARE - SOE							1		
						1				2	C91-P623-3-14-15.16	
07	AAAXC99		SPARE - SOE							1		
						1				2	C91-P623-3-15-1.2	
12	CSHBC14		HPCS PUMP BREAKER NO 2		OPEN		1	0		1	H13-P630-C	S2/A-2CSHH10
	N/A	N/A	7.243-001-011	0 0	CLOSED	1				1	C91-P623-4-00-1.2	
12	CSHBC15		HPCS PRESSURE		LOW		0	0		1	H13-P630-C	E22A-K51
	N/A	N/A	7.243-001-011	0 0	NORMAL	1				1	C91-P623-4-00-3.4	
12	CSHBC16		HPCS INJECTION FLOW		NORMAL		1	0		1	H13-P630-C	E22A-K56
	N/A	N/A	7.243-001-011	0 0	HIGH	1				1	C91-P623-4-00-5.6	

ATTACHMENT 11

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM 21HC- 4
CONTROLLED ISSUE - REVISION 9

R V	PT ID	POINT DESCRIPTION	PR-0	A S	CO USED	CUT-OUT 1 POINT ID	COI IS	SI TS	SOURCE	TROUBLE CONTACT
	ESH	LSH	VENDOR REF DWG	P T / N T	PR-1	S E	ALM NO	ANN WIND	VO LT	DESTINATION
14	AAABC79	SPARE - SOE		1				1	1 C91-P623-3-07-15,16	
14	AAABC80	SPARE - SOE		1				1	1 C91-P623-3-07-5,6	
14	AAABC81	SPARE - SOE		1				1	1 C91-P623-3-07-7,8	
15	AAABC82	SPARE - SOE		1				1	1 C91-P623-4-15-9,10	
15	AAABC83	SPARE - SOE		1				1	1 C91-P623-4-15-11,12	
15	AAABC84	SPARE - SOE		1				1	1 C91-P623-4-15-13,14	
15	AAABC85	SPARE - SOE		1				1	1 C91-P623-4-15-15,16	
13	AAABC86	SPARE - SOE		1				1	1 C91-P623-3-01-1,2	
13	AAABC87	SPARE - SOE		1				1	1 C91-P623-3-01-3,4	
13	AAABC88	SPARE - SOE		1				1	1 C91-P623-3-01-5,6	
12	AAABC89	SPARE - SOE		1				1	1 C91-P623-3-01-7,8	
09	AAABC90	SPARE - SOE		1				1	1 C91-P623-3-01-9,10	
09	AAABC91	SPARE - SOE		1				1	1 C91-P623-3-01-11,12	
09	AAABC92	SPARE - SOE		1				1	1 C91-P623-3-01-13,14	
09	AAABC93	SPARE - SOE		1				1	1 C91-P623-3-01-15,16	
09	AAABC94	SPARE - SOE		1				1	1 C91-P623-3-02-1,2	
09	AAABC95	SPARE - SOE		1				1	1 C91-P623-3-02-3,4	
09	AAABC96	SPARE - SOE		1				1	1 C91-P623-3-02-5,6	
09	AAABC97	SPARE - SOE		1				1	1 C91-P623-3-02-7,8	
09	AAABC98	SPARE - SOE		1				1	1 C91-P623-3-02-9,10	

8/85

ATTACHMENT 1J

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM 2IHC- 6
CONTROLLED ISSUE - REVISION 9

R V	PT ID / ESH	LSH	POINT DESCRIPTION / VENDOR REF DWG	PR-0 / P T / N T / PR-1	A S / S E	CO USED / ALM NO	CUT-OUT 1 POINT 10 / ALM MNMO	COL IS	SI TS / VO LT		SOURCE / DESTINATION	TROUBLE CONTACT
	09 AAABC99		SPARE - SOE		1				1	1	C91-P623-3-02-11,12	
	12 AAADC26		SPARE - SOE		1				1	1	C91-P623-4-03-13,14	
	12 AAADC27		SPARE - SOE		1				1	1	C91-P623-4-03-15,16	
	12 AAADC28		SPARE - SOE		1				1	1	C91-P623-4-12-15,16	
	12 AAADC29		SPARE - SOE		1				1	1	C91-P623-4-13-15,16	
	12 AAADC30		SPARE - SOE		1				1	1	C91-P623-4-14-13,14	
	12 AAADC31		SPARE - SOE		1				1	1	C91-P623-4-14-15,16	
	15 AAAUC01		SPARE - SOE		1				2	1	C91-P623-3-15-3,4	
	15 AAAUC02		SPARE - SOE		1				2	1	C91-P623-3-15-5,6	
	15 AAAUC03		SPARE - SOE		1				2	1	C91-P623-3-15-7,8	
	15 AAAUC04		SPARE - SOE		1				2	1	C91-P623-3-15-9,10	
	15 AAAUC05		SPARE - SOE		1				2	1	C91-P623-3-15-11,12	
	15 AAAUC06		SPARE - SOE		1				2	1	C91-P623-3-15-13,14	
	15 AAAUC07		SPARE - SOE		1				2	1	C91-P623-3-15-15,16	
	09 AAAXC01		SPARE - SOE		1				1	1	C91-P623-3-02-13,14	
	09 AAAXC02		SPARE - SOE		1				1	1	C91-P623-3-02-15,16	
	07 AAAXC03		SPARE - SOE		1				1	1	C91-P623-3-03-1,2	
	07 AAAXC04		SPARE - SOE		1				1	1	C91-P623-3-03-3,4	
	07 AAAXC05		SPARE - SOE		1				1	1	C91-P623-3-03-5,6	
	07 AAAXC06		SPARE - SOE		1				1	1	C91-P623-3-03-7,8	

ATTACHMENT 1K

03/08/85

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM 2IIC- 6
CONTROLLED ISSUE - REVISION 9

R Y	PT ID	POINT DESCRIPTION	PR-0	A S	CO USED	CUT-OUT 1 POINT ID	CO1 IS	SI IS	SOURCE	TROUBLE CONTACT
	ESH	LSH	VENDOR REF DWG	P T / N T	PR-1	S E	ALM NO	ANN WIND	VO LT	DESTINATION

PT ID	POINT DESCRIPTION	A S / S E	SI IS	VO LT	DESTINATION
12 AAAXC07	SPARE - SOE	1	1	1	C91-P623-3-03-9,10
12 AAAXC08	SPARE - SOE	1	1	1	C91-P623-3-03-11,12
12 AAAXC09	SPARE - SOE	1	1	1	C91-P623-3-03-13,14
12 AAAXC10	SPARE - SOE	1	1	1	C91-P623-3-03-15,16
15 AAAXC11	SPARE - SOE	1	1	1	C91-P623-3-04-1,2
15 AAAXC12	SPARE - SOE	1	1	1	C91-P623-3-04-3,4
15 AAAXC13	SPARE - SOE	1	1	1	C91-P623-3-04-5,6
15 AAAXC15	SPARE - SOE	1	1	1	C91-P623-3-04-9,10
15 AAAXC16	SPARE - SOE	1	1	1	C91-P623-3-04-11,12
15 AAAXC17	SPARE - SOE	1	1	1	C91-P623-3-04-13,14
15 AAAXC18	SPARE - SOE	1	1	1	C91-P623-3-04-15,16
15 AAAXC19	SPARE - SOE	1	1	1	C91-P623-3-05-1,2
15 AAAXC20	SPARE - SOE	1	1	1	C91-P623-3-05-3,4
15 AAAXC21	SPARE - SOE	1	1	1	C91-P623-3-05-5,6
15 AAAXC22	SPARE - SOE	1	1	1	C91-P623-3-05-7,8
15 AAAXC23	SPARE - SOE	1	1	1	C91-P623-3-05-9,10
15 AAAXC24	SPARE - SOE	1	1	1	C91-P623-3-05-11,12
15 AAAXC25	SPARE - SOE	1	1	1	C91-P623-3-05-13,14
15 AAAXC26	SPARE - SOE	1	1	1	C91-P623-3-05-15,16
15 AAAXC27	SPARE - SOE	1	1	1	C91-P623-3-06-1,2

ATTACHMENT 1L

03/08/05

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM 2INC- 6
CONTROLLED ISSUE - REVISION 9

R V	PT ID	POINT DESCRIPTION	PR-0	A S	CO USED	CUT-OUT 1 POINT ID	COL IS	SI IS		SOURCE	TROUBLE CONTACT
	ESK	LSK / VENDOR REF DWG / P T N T	PR-1	S E	ALM NO	ANN WIND		VO LT		DESTINATION	
15	AAAXC28	SPARE - SOE		1				1	1	C91-P623-3-06-3,4	
13	AAAXC29	SPARE - SOE		1				1	1	C91-P623-3-06-5,6	
13	AAAXC30	SPARE - SOE		1				1	1	C91-P623-3-06-7,8	
13	AAAXC31	SPARE - SOE		1				1	1	C91-P623-3-06-9,10	
13	AAAXC32	SPARE - SOE		1				1	1	C91-P623-3-06-11,12	
13	AAAXC33	SPARE - SOE		1				1	1	C91-P623-3-06-13,14	
13	AAAXC34	SPARE - SOE		1				1	1	C91-P623-3-06-15,16	
14	AAAXC35	SPARE - SOE		1				1	1	C91-P623-3-07-1,2	
14	AAAXC36	SPARE - SOE		1				1	1	C91-P623-3-07-3,4	
07	AAAXC39	SPARE - SOE		1				1	1	C91-P623-3-07-9,10	
07	AAAXC40	SPARE - SOE		1				1	1	C91-P623-3-07-11,12	
07	AAAXC41	SPARE - SOE		1				1	1	C91-P623-3-07-13,14	
07	AAAXC43	SPARE - SOE		1				2	1	C91-P623-3-08-1,2	
07	AAAXC44	SPARE - SOE		1				2	1	C91-P623-3-08-3,4	
07	AAAXC45	SPARE - SOE		1				2	1	C91-P623-3-08-5,6	
07	AAAXC46	SPARE - SOE		1				2	1	C91-P623-3-08-7,8	
07	AAAXC47	SPARE - SOE		1				2	1	C91-P623-3-08-9,10	
07	AAAXC48	SPARE - SOE		1				2	1	C91-P623-3-08-11,12	
07	AAAXC49	SPARE - SOE		1				2	1	C91-P623-3-08-13,14	
07	AAAXC50	SPARE - SOE		1				2	1	C91-P623-3-08-15,16	

ATTACHMENT 1M

03/08/85

NINE MILE POINT - UNIT 2
COMPUTER POINTS I/O LIST
SOE REPORT
COMPUTER SHEETS FOR LOOP DIAGRAM ZINC- 6
CONTROLLED ISSUE - REVISION 9

RV	PT ID	POINT DESCRIPTION	PR-M	AS / SE	CO USED / ALM NO	CUT-OUT 1 POINT ID / ANN WIND	COL IS	SI TS / VO LT	SOURCE / DESTINATION	TROUBLE CONTACT
07	AAAXC51	SPARE - SOE		1				2	1 / C91-P623-3-09-1,2	
07	AAAXC52	SPARE - SOE		1				2	1 / C91-P623-3-09-3,4	
07	AAAXC53	SPARE - SOE		1				2	1 / C91-P623-3-09-5,6	
07	AAAXC54	SPARE - SOE		1				2	1 / C91-P623-3-09-7,8	
07	AAAXC55	SPARE - SOE		1				2	1 / C91-P623-3-09-9,10	
07	AAAXC56	SPARE - SOE		1				2	1 / C91-P623-3-09-11,12	
07	AAAXC57	SPARE - SOE		1				2	1 / C91-P623-3-09-13,14	
07	AAAXC58	SPARE - SOE		1				2	1 / C91-P623-3-09-15,16	
07	AAAXC59	SPARE - SOE		1				2	1 / C91-P623-3-10-1,2	
07	AAAXC60	SPARE - SOE		1				2	1 / C91-P623-3-10-3,4	
07	AAAXC61	SPARE - SOE		1				2	1 / C91-P623-3-10 5,6	
07	AAAXC62	SPARE - SOE		1				2	1 / C91-P623-3-10 7,8	
07	AAAXC63	SPARE - SOE		1				2	1 / C91-P623-3-10-9,10	
07	AAAXC64	SPARE - SOE		1				2	1 / C91-P623-3-10-11,12	
07	AAAXC65	SPARE - SOE		1				2	1 / C91-P623-3-10-13,14	
07	AAAXC66	SPARE - SOE		1				2	1 / C91-P623-3-10-15,16	
07	AAAXC67	SPARE - SOE		1				2	1 / C91-P623-3-11-1,2	
07	AAAXC68	SPARE - SOE		1				2	1 / C91-P623-3-11-3,4	
07	AAAXC69	SPARE - SOE		1				2	1 / C91-P623-3-11-5,6	
07	AAAXC70	SPARE - SOE		1				2	1 / C91-P623-3-11-7,8	

Second header row: ESK, LSK, VENDOR REF DWG, P T H T, PR-1, S E, ALM NO, ANN WIND, VO LT, DESTINATION

UNIT X, PAGE XX OF XX MO/DA/YR HR:MN

SEQUENCE OF EVENTS LOG

TIME	PT ID	NAME	STATUS
HR:MN:SC.MMM	PPPPPPPP	AAAAAAAAAAAAAAAAAAAAAAAAAA	AAAAAAAA
HR:MN:SC.MMM	PPPPPPPP	AAAAAAAAAAAAAAAAAAAAAAAAAA	AAAAAAAA
HR:MN:SC.MMM	PPPPPPPP	AAAAAAAAAAAAAAAAAAAAAAAAAA	AAAAAAAA
HR:MN:SC.MMM	PPPPPPPP	AAAAAAAAAAAAAAAAAAAAAAAAAA	AAAAAAAA
HR:MN:SC.MMM	PPPPPPPP	AAAAAAAAAAAAAAAAAAAAAAAAAA	AAAAAAAA
HR:MN:SC.MMM	PPPPPPPP	AAAAAAAAAAAAAAAAAAAAAAAAAA	AAAAAAAA

RECORD INITIATED AFTER 64 CONTACT CHANGES FOR 30 SECONDS
AFTER FIRST CONTACT CHANGE, WHICHEVER IS FIRST.

Figure 3-10. SEQ-1 Sequence of Events Log

```
UNIT 1 , PAGE II OF II                                        MO-DA-YR  HR:MN
      HISTORICAL DATA RETRIEVAL AND REVIEW SERVICES (ONLINE)

        BETWEEN HR:MN ON MM/DD/YY AND HR:MN ON MM/DD/YY
                       SEQUENCE OF EVENTS LOG

  TIME    POINT ID POINT DESCRIPTION         STATUS
HR:MN:SC PPPPPPPP DDDDDDDDDDDDDDDDDDDDDDDD SSS
```

TO DISPLAY MORE INFORMATION, PRESS PAGE FORWARD

FILE DHR3-1.BWR/6

DHR3-1 - Historical Data Retrieval and Review Services - SEQUENCE OF EVENTS LOG.

ATTACHMENT 4

COMPUTER POINTS

For NSSS Post Trip Log

FWSLA101	Reactor Water Level
RCSFB01	Recirc Flow Total
FWSB01	Feedwater Flow Total
FWSFA103	Main Steam Flow Total
FWSFA104	Main Turbine Steam Flow
NSSPB01	Reactor Pressure
FWSPA101	Steam Dome Pressure
CNMFB02	Cond Booster Flow
CNMFB01	Cond Pump Flow
RCSTA103	Recirc Pump Suct Temp A
RCSTA105	Recirc Pump Suct Temp B
CMSPA01	Drywell Pressures
CMSTA01	Drywell Temp
CMSPA02	Drywell High Range Pressure
CMSPA04	Suppression Pool Pressure
CMSTA07	Suppression Pool Temp
CMSAA02	Drywell Oxygen
MSSBC20	Bypass Valve Position
CWSTB10	Avg Cond Temp Rise
SWPTA53	Service Water Inlet °F
SWPTA74	Service Water Disch °F
CNSLA03	Hotwell Level
TMLPA02	Turbing Big Oil Press
CNMPA02	Cond Vac

ATTACHMENT 5

COMPUTER POINTS

For BOP Post Trip Log

NMP2A273	APRM A
NMP2A274	APRM B
NMP2A275	APRM C
NMP2A276	APRM D
NMP2A277	APRM E
NMP2A278	APRM F
CNMPA04	Condensate Pumps Discharge Header Pressure
FWSPA04	Final Feedwater Pressure To Reactor
CNMPA01	Condenser Vacuum 1A
MSSPA05	Turbine Main Steam Inlet Hdr. Pressure
MSSPA06	Turbine 1st Stage Pressure
SPGQA02	Generator Water
SWPFA08	Service Water Pump Loop B Hdr. Flow
SWPFA09	SWP Loop A Header Flow
SWPFPA15	Service Water Loop A Disch Pressure
SWPFPA16	Service Water Loop B Disch Pressure
TMBPA01	Hydraulic Fluid Pressure
TMEPA01	Gland Seal Steam Supply Pressure
FWSPA100	Reactor Pressure
MSSFA101	Cleanup Flow
NMPFA101	Recirc Loop A1 Drive Flow
NMPFA103	Recirc Loop B1 Drive Flow
NSSFA101	Total Care Flow
RCSTA103	Recirc Loop A1 Inlet Temp
RCSTA105	Recirc Loop B1 Inlet Temp
CCPPA01	RBCLCW Pump Disch Hdr. Press
CCPTA16	RBCLCW Heat Exchange Disch Temp
CCSPA01	TBCLCW Pump Disch Hdr. Press
CNMPA03	Condenser Vacuum 1B
HVRPA01	Reactor Bldg. Differential Pressure
MSSTA03	Turb PSV89A Outlet Temperature
OFGFA01	Offgas System Total Flow
OFGPA01	Offgas System Inlet Pressure
TMEPA03	Clean Steam Reboiler E1A Disch Steam Pressure
TMEPA04	Cleam Steam Reboiler E1B Disch Steam Pressure

```
UNIT X,PAGE XX OF XX                              DATE MA/DY/YR TIME 10:30:10

                     NSS POST TRIP REVIEW LOG

                         PRE-TRIP DATA

         PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP
HR.
10:20:10 SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX
10:20:15 SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX
10:20:20
10:20:25
10:20:30

                         POST-TRIP DATA

         PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP PPPPPPPP

10:25:10 SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX
10:25:15 SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX SXXXX.XX
10:25:25
10:25:30

          DATA RECORDED FOR 24 POINTS AT 5-SEC INTERVALS FOR 5 MIN

NOTE:BOP POST TRIP REVIEW LOG IS SAME AS NSS LOG EXCEPT,BOP LOG DATA IS RECORDED FOR
    48 POINTS AT 15 SEC INTERVALS FOR 30 MINUTES BEFOR AND 30 MIN AFTER TRIP.
```

Figure 3-6. PTL-2 NSS Post Trip Review Log

Section 1.2B

Generic Letter 83-28

Post-Trip Review - Data and Information Capability

General Electric's Transient Analysis Recording System

GETARS - 1

1.2.1 Capability for assessing sequency of events (on-off indications).

1.2.1.1 Brief description of equipment (e.g., plant computer, dedicated computer, strip chart).

NMP2 Response

The General Electric Transient Analysis Recording System (GETARS-1) is a high-speed data acquisition system developed for startup operations but is a permanent plant system. The system operates on a Hewlett-Packard 2117F computer system. The processor contains 128K words of high-speed memory, dual-channel direct memory access, and a dynamic mapping system. The system utilizes a Hewlett-Packard 7920 moving head disk, which has a capability of 50 mbytes for program and data storage. A Hewlett-Packard 7970E magnetic tape drive is used for historical recording. The system also utilizes the Validyne HD310 Expanded Multiplexer system as the analog to digital converter. This system can contain up to 4096 analog inputs.

Peripherals contained on the system include the Versatec V80 printer/plotter, and one HP2645A black and white video display.

The operating system uses two sets of supervisory software. The realtime executive system is the RTE-IVB. This system executes all data entry programs, data reduction programs, and utility programs. A lower overhead executive called RTEM is used to permit interfacing between peripheral devices and (control for high-speed data) acquisition programs.

These are also 21 permanent remote multiplexes used for analog scanning.

1.2.1.2 Parameters monitored.

NMP2 Response

The GETARS system presently contains approximately 500 Analog points. Attachment 1 contains a list of the systems and the ID points which are monitored by the GETARS system.

1.2.1.3 Time discrimination between events.

NMP2 Response

The remote multiplexers (MC370AD) each contain 32 analog channels. The scan rates range from 23,810 scans/second when monitoring one channel, to 2,100 scans/second when monitoring all 32 channels. In a real time environment, groups may not be scanned more rapidly than once each millisecond i.e., 1,000 samples per second.

1.2.1.4 Format for displaying data and information.

NMP2 Response

There are a number of functions contained on the GETARS system that produce various reports and plots.

The control rod timing function indentifies the status of each control rod and evaluates control rod scram performance against test time criteria. Attachments 2 through 7 provide format examples for the reports generated by this function.

The off-line Print/Plot program provides on-site verification and analysis of data recorded by the data acquisition system. Attachments 8 and 9 represent the format associated with the printer and the plotter.

The dynamic noise frequency analysis function is a time series analysis package which allows time history data to be analyzed in the frequency domain. Attachment 10 provides this function's output format.

The histogram function provides a display of signal data in either engineering units, millivolts, or engineering units. Attachment 12 provides a sample of this function's output.

The Run analysis function provides a statistical analysis of a given data acquisition Run. A sample format is provided on Attachment 13.

1.2.1.5 Capability for retention of data and information.

NMP2 Response

Data retention for the GETARS system is contained on either the Disk Drive or Magnetic Tape. System utilities are available on this system to save the data to or from tape. The data acquisition system automatically writes analog data to the disk.

1.2.1.6 Power source(s) (e.g., Class 1E, non-Class 1E, noninterruptible).

NMP2 Response

Power to the General Electric Transient Analysis Response System (GETARS) is supplied by an Uninterruptible Power Supply 2VBB-UPS1G Non-Class 1E. This supply is fed from a 600V power panel 2VBB-PNL301, which is supplied by one of two sources, either the Station Generator 13.8KV line (2NJS-US3, during normal operation) or from an off-site Scriba 115KV line (2NJS-US4, during a shutdown condition). The GETARS system is also supplied by an alternate 600V BUS 2NJS-US6. In a condition by which all power is lost, backup power is supplied by a 125V DC battery supply 2BYS-SWG001C.

1.2.1.6 NMP2 Response (Cont'd)

In summary, upon loss of normal power, a static transfer switch transfers power from the normal source to the alternative source. If both normal and alternate sources are lost, the DC source will automatically pickup the loads (by means of a DC auctioneering circuit) and supply power panel 2VBS-PNLC102 which feeds GETARS.

1.2.2 Capability for assessing the time history of analog variables needed to determine the cause of unscheduled reactor shutdowns and the functioning of safety-related equipment.

1.2.2.1 Brief description of equipment (e.g., plant computer, dedicated computer, strip charts).

NMP2 Response

A description of the equipment making up the GETARS system is provided in the response for Section 1.2.1.1.

1.2.2.2 Parameters monitored, sampling rate and basis for selecting parameters and sampling rate.

NMP2 Response

All system inputs contained in the systems described on Attachment 1 are continually being monitored. As stated in Section 1.2.1.3, the absolute maximum recording speed is 1,000 samples per channel.

1.2.2.3 Duration of time history (minutes before trip and minutes after trip).

NMP2 Response

Upon a trip condition, data recording continues until the disk data area becomes full or the operator terminates the data recording. This disk area will hold a maximum of 11 minutes of data of which one-sixth is pre-trip data.

1.2.2.4 Format for displaying data including scale (readability) of time histories.

NMP2 Response

Description of the formats for displaying the recorded data is contained in Section 1.2.1.4.

1.2.2.5 Capability for retention of data, informatin and physical evidence (both hardware and software).

NMP2 Response

Description of the capability for retention of data is contained in Section 1.2.1.5.

1.2.2.6 Power source(s) (e.g., Class 1E, non-Class 1E, noninterruptible).

NMP2 Response

A description of the power sources is contained in Section 1.2.1.6.

1.2.3 Other data and information provided to assess the cause of unscheduled reactor shutdowns.

1.2.4 Schedule for any planned changes to existing data and information capability.

NMP2 Response

See section 1.2.4.A.

Attachment 1

Unit 2's GETAR's System (parameters Monitored) and ID Points

Main Steam

Steam Line Flow
Main Steam Header Pressure
Main Steam Line Isolation
MSIV Position

RPS

Manual Reactor Scram
Auto Reactor Scram

Rx Instrumentation

Rx Dome Pressure
Rx Water Level
Rx Core Plate DP
Rx Bottom Head Drain Temperature
Rx Vessel Level (WR)

Neutron Monitoring

APRM's
LPRM's
Thermal Heat Flux
Flow-Biased Thermal Upscale Trip Setpoint

Residual Heat Removal

RHR Hx Level
RHR Sys Flow
RHR Hx Lvl Cont Output
RHR Hx Pressure
RHR Pump Trp Brkr. Posn.
CRD System Flow
Selectable CRD Position

Attachment 1 (Cont'd)

Recirculation System

Recirc Loop Flow
Recirc Loop Flow Control Valve Position
Recirc Pmp Trip Bkr
Recirc Master Controller Output
Load Demand Error
Recirc Sys Flux Error
Recirc Sys Flux Estimator Output
Recirc Loop Suction Temp
Recirc Pump D/P
Jet Pump Double Tap D/P
Jet Pump Flow Loops
Total Core Flow
LFMG Drive Motor Bkr
Recirc Flow Control Funct Generator Inputs

RCIC

RCIC Initiation
RCIC Suction Pressure Controller Output
RCIC Elbow Tap D/P
RCIC Turbine Spd
RCIC Flow

Feedwater

Fdwtr Line Temp
Fdwtr Flow
Stm/Fdwtr Flow Mismatch
Fdwtr Master Controller Output
Fdwtr Pump Suction Pressure
Fdwtr Pump Disch Pressure
Fdwtr Pump Byp Low Flow Control Valve Pos
Fdwtr Pump Trip
Fdwtr Low Flow Valve Pos
Fdwtr High Flow Control Valve Pos
Fdwtr Recirc Valve Pos
High Flow Funct Generator Output
Low Flow Master Controller Level Setpoint
STep Generator Output

HPCS/LPCS

HPCS Pump Trip
HPCS Initiation
HPCS Discharge Flow
HPCS Discharge Pressure
HPCS Diesel Generator Bkr Trip
HPCS MCC Feeder Bkr Trip
LPCS Injection Valve Pos

Attachment 1 (Cont'd)

Control Rod Drive

CRD Flow Controller Output
RCIC Trip/Throttle Valve Pos

Safety Relief Valve

SRV's
ADS Initiation

BOP/Emergency Bus Breaker

Normal Brkr Pos
Alternate Brkr Pos
Diesel Brkr Pos
MCC Feeder Breaker Pos

BOP (Turbine/Generator)

Main Turbine Speed
Auto Load Following
Load Reference Output
Main Turbine trip
Transient Auto Pressure Setpoint
EHC Pressure Setpoint
Power/Load Unbalance
Stop Valve Pos
Bypass Valve Pos
Main Turbine Stm Flow
Main Generator MWE
Grid Voltage
Grid Frequency
RCIC Ramp Gen Signal Converter Ouput
RCIC EGM Output
RCIC Steam Control (Governor) Valve Pos
Total Byp Valve Posn

BOP (Condenser, Extract Stm, Service Water, FW Heater

Cond. Bstr Pmp Disch Hdr Press
Service Water Pump Trips
Spent Fuel Pool Cooling Pump Trip
Service Water Pump Trip
Main Condenser Vacuum
Cond Pump Disch Hdr Press
Htr Drn Pmp Disch Press
Lp Htr Strings
A&B Isolation Valves
Bypass Rx FWP Bypass
ESS LP/HP Htr Strings
Warming Valves
Main Gen Bkr Pos
Press Reg Output
Total Cont Valve Pos

4.2.3 Sample Output - Channel And Subchannel Number

CRDED OUTPUT 12 22 AM THU., 26 JUNE, 1984

CRD I.D. CHANNEL/SUBCHANNEL NUMBER

3															
61					336/ 1	0/ 0	0/ 0	373/ 1	0/ 0	371/ 1	0/ 0				
57			336/ [illegible]	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	336/ 3	0/ 0	0/ 0	0/ 0		
53		0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	
49		336/ 4	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	
45	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0
41	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0
37	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	371/ 3
33	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0
29	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0
25	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0
21	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0
17		0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	
13		0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	
9			0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0	0/ 0		
5					0/ 0	0/ 0	371/ 2	0/ 0	0/ 0	0/ 0	0/ 0				
1	4	8	12	16	20	24	28	32	36	40	44	48	52	56	60

Sample Output Con't - Channel and Subchannel Number

CRD I.D. STATUS OF ALL CRD'S

J															
61					1	Ø	Ø	3	Ø	2	Ø				
57			Ø	Ø	Ø	Ø	Ø	Ø	Ø	4	Ø	Ø	Ø		
53		Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	
49		99	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	
45	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
41	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
37	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	99
33	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
29	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
25	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
21	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
17		Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	
13		Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	
9			Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø		
5					Ø	Ø	99	Ø	Ø	Ø	Ø				
1	4	8	12	16	20	24	28	32	36	40	44	48	52	56	60

Ø => NO DATA, 1 => FAST ROD, 2 => NORMAL ROD, 3 => SLOW ROD, 4 => FAILED ROD, 99 => INOPERATIVE ROD

ATTACHMENT 4

4.2.4 Sample Output - Rod Timming

CRD #	STATUS	DATE	TIME	CRD #	STATUS	DATE	TIME	CRD #	STATUS	DATE	TIME
20.61	1	4/28/81	3: 0	24.61	0	6/28/84	0:10	28.61	0	6/28/84	0:10
32.61	3	12/12/80	0:30	36.61	0	6/28/84	0:10	40.61	2	1/18/81	0:40
44.61	0	6/28/84	0:10	12.57	0	4/28/81	1:20	16.57	0	6/28/84	0:10
20.57	0	6/28/84	0:10	24.57	0	6/28/84	0:10	28.57	0	6/28/84	0:10
32.57	0	6/28/84	0:10	36.57	0	6/28/84	0:10	40.57	4	4/25/81	3:20
44.57	0	6/28/84	0:10	48.57	0	6/28/84	0:10	52.57	0	6/28/84	0:10
8.53	0	6/28/84	0:10	12.53	0	6/28/84	0:10	16.53	0	6/28/84	0:10
20.53	0	6/28/84	0:10	24.53	0	6/28/84	0:10	28.53	0	6/28/84	0:10
32.53	0	6/28/84	0:10	36.53	0	6/28/84	0:10	40.53	0	6/28/84	0:10
44.53	0	6/28/84	0:10	48.53	0	6/28/84	0:10	52.53	0	6/28/84	0:10
56.53	0	6/28/84	0:10	8.49	99	4/23/81	6:12	12.49	0	6/28/84	0:10
16.49	0	6/28/84	0:10	20.49	0	6/28/84	0:10	24.49	0	6/28/84	0:10
28.49	0	6/28/84	0:10	32.49	0	6/28/84	0:10	36.49	0	6/28/84	0:10
40.49	0	6/28/84	0:10	44.49	0	6/28/84	0:10	48.49	0	6/28/84	0:10
52.49	0	6/28/84	0:10	56.49	0	6/28/84	0:10	4.45	0	6/28/84	0:10
8.45	0	6/28/84	0:10	12.45	0	6/28/84	0:10	16.45	0	6/28/84	0:10
20.45	0	6/28/84	0:10	24.45	0	6/28/84	0:10	28.45	0	6/28/84	0:10
32.45	0	6/28/84	0:10	36.45	0	6/28/84	0:10	40.45	0	6/28/84	0:10
44.45	0	6/28/84	0:10	48.45	0	6/28/84	0:10	52.45	0	6/28/84	0:10
56.45	0	6/28/84	0:10	60.45	0	6/28/84	0:10	4.41	0	6/28/84	0:10
8.41	0	6/28/84	0:10	12.41	0	6/28/84	0:10	16.41	0	6/28/84	0:10
20.41	0	6/28/84	0:10	24.41	0	6/28/84	0:10	28.41	0	6/28/84	0:10
32.41	0	6/28/84	0:10	36.41	0	6/28/84	0:10	40.41	0	6/28/84	0:10
44.41	0	6/28/84	0:10	48.41	0	6/28/84	0:10	52.41	0	6/28/84	0:10
56.41	0	6/28/84	0:10	60.41	0	6/28/84	0:10	4.37	0	6/28/84	0:10
8.37	0	6/28/84	0:10	12.37	0	6/28/84	0:10	16.37	0	6/28/84	0:10
20.37	0	6/28/84	0:10	24.37	0	6/28/84	0:10	28.37	0	6/28/84	0:10
32.37	0	6/28/84	0:10	36.37	0	6/28/84	0:10	40.37	0	6/28/84	0:10
44.37	0	6/28/84	0:10	48.37	0	6/28/84	0:10	52.37	0	6/28/84	0:10
56.37	0	6/28/84	0:10	60.37	99	12/12/81	23: 0	4.33	0	6/28/84	0:10
8.33	0	6/28/84	0:10	12.33	0	6/28/84	0:10	16.33	0	6/28/84	0:10
20.33	0	6/28/84	0:10	24.33	0	6/28/84	0:10	28.33	0	6/28/84	0:10
32.33	0	6/28/84	0:10	36.33	0	6/28/84	0:10	40.33	0	6/28/84	0:10
44.33	0	6/28/84	0:10	48.33	0	6/28/84	0:10	52.33	0	6/28/84	0:10
56.33	0	6/28/84	0:10	60.33	0	6/28/84	0:10	4.29	0	6/28/84	0:10
8.29	0	6/28/84	0:10	12.29	0	6/28/84	0:10	16.29	0	6/28/84	0:10
20.29	0	6/28/84	0:10	24.29	0	6/28/84	0:10	28.29	0	6/28/84	0:10
32.29	0	6/28/84	0:10	36.29	0	6/28/84	0:10	40.29	0	6/28/84	0:10
44.29	0	6/28/84	0:10	48.29	0	6/28/84	0:10	52.29	0	6/28/84	0:10
56.29	0	6/28/84	0:10	60.29	0	6/28/84	0:10	4.25	0	6/28/84	0:10
8.25	0	6/28/84	0:10	12.25	0	6/28/84	0:10	16.25	0	6/28/84	0:10
20.25	0	6/28/84	0:10	24.25	0	6/28/84	0:10	28.25	0	6/28/84	0:10
32.25	0	6/28/84	0:10	36.25	0	6/28/84	0:10	40.25	0	6/28/84	0:10
44.25	0	6/28/84	0:10	48.25	0	6/28/84	0:10	52.25	0	6/28/84	0:10
56.25	0	6/28/84	0:10	60.25	0	6/28/84	0:10	4.21	0	6/28/84	0:10
8.21	0	6/28/84	0:10	12.21	0	6/28/84	0:10	16.21	0	6/28/84	0:10
20.21	0	6/28/84	0:10	24.21	0	6/28/84	0:10	28.21	0	6/28/84	0:10
32.21	0	6/28/84	0:10	36.21	0	6/28/84	0:10	40.21	0	6/28/84	0:10
44.21	0	6/28/84	0:10	48.21	0	6/28/84	0:10	52.21	0	6/28/84	0:10
56.21	0	6/28/84	0:10	60.21	0	6/28/84	0:10	8.17	0	6/28/84	0:10
12.17	0	6/28/84	0:10	16.17	0	6/28/84	0:10	20.17	0	6/28/84	0:10
24.17	0	6/28/84	0:10	28.17	0	6/28/84	0:10	32.17	0	6/28/84	0:10
36.17	0	6/28/84	0:10	40.17	0	6/28/84	0:10	44.17	0	6/28/84	0:10
48.17	0	6/28/84	0:10	52.17	0	6/28/84	0:10	56.17	0	6/28/84	0:10
8.13	0	6/28/84	0:10	12.13	0	6/28/84	0:10	16.13	0	6/28/84	0:10
20.13	0	6/28/84	0:10	24.13	0	6/28/84	0:10	28.13	0	6/28/84	0:10
32.13	0	6/28/84	0:10	36.13	0	6/28/84	0:10	40.13	0	6/28/84	0:10
44.13	0	6/28/84	0:10	48.13	0	6/28/84	0:10	52.13	0	6/28/84	0:10
56.13	0	6/28/84	0:10	12. 9	0	6/28/84	0:10	16. 9	0	6/28/84	0:10
20. 9	0	6/28/84	0:10	24. 9	0	6/28/84	0:10	28. 9	0	6/28/84	0:10
32. 9	0	6/28/84	0:10	36. 9	0	6/28/84	0:10	40. 9	0	6/28/84	0:10
44. 9	0	6/28/84	0:10	48. 9	0	6/28/84	0:10	52. 9	0	6/28/84	0:10
20. 5	0	6/28/84	0:10	24. 5	0	6/28/84	0:10	28. 5	99	1/ 1/82	0: 0
32. 5	0	6/28/84	0:10	36. 5	0	6/28/84	0:10	40. 5	0	6/28/84	0:10
44. 5	0	6/28/84	0:10								

ATTACHMENT 5

4.2.5 Sample Input - CRDSV

```
:RU,CRDSV
 LIST DEVICE LU?                                                76
 THIS PROGRAM WILL CHECK SCRAM DATES AGAINST SCRAM
 SURVEILLANCE REQUIREMENTS AND WILL PRINTOUT THE RESULTS
 TO HELP THE USER IN IDENTIFYING RODS DUE FOR TESTING.
 INPUT FILE NAME?                                                ?CRD#1
```

4.2.6 Sample Output - CRDSV

CRDSV OUTPUT 12:27 AM THU. 28 JUNE, 1984

ALL RODS HAVE BEEN SCRAM TESTED WITHIN 1294 DAYS.

***** SURVEILLANCE INTERVALS HAVE BEEN EXCEEDED *****

THE FOLLOWING RODS HAVE SCRAM TIME DATES LESS THAN 120 DAYS

DATE	ROD COORDINATE	STATUS

THE FOLLOWING 20% OF THE RODS HAVE THE OLDEST SCRAM TEST DATES

DATE	ROD COORDINATE	STATUS
12/12/80	32,61	3
1/18/81	40,61	2
4/28/81	20,61	1
6/28/84	24,61	0
6/28/84	36,61	0
6/28/84	28,61	0
6/28/84	44,61	0
4/28/81	12,57	0
6/28/84	16,57	0
6/28/84	20,57	0
6/28/84	24,57	0
6/28/84	28,57	0
6/28/84	32,57	0
6/28/84	36,57	0
4/25/81	40,57	4
6/28/84	44,57	0
6/28/84	48,57	0
6/28/84	52,57	0
6/28/84	8,53	0
6/28/84	12,53	0
6/28/84	16,53	0
6/28/84	20,53	0
6/28/84	24,53	0
6/28/84	28,53	0
6/28/84	32,53	0
6/28/84	36,53	0
6/28/84	40,53	0
6/28/84	44,53	0
6/28/84	48,53	0
6/28/84	52,53	0
6/28/84	56,53	0
4/23/81	8,49	99
6/28/84	12,49	0
6/28/84	16,49	0
6/28/84	20,49	0
6/28/84	24,49	0
6/28/84	28,49	0
6/28/84	32,49	0

4.2.9 Sample Output - List Of Rods

LIST OF RODS WITH STATUS 4 , 99 & 0

RODS WHICH HAVE FAILED SCRAM TIME REQUIREMENTS STATUS 4

ROD 40.57

OPERATOR DECLARED INOPERATIVE RODS" STATUS 99

ROD 8.49
ROD 60.37
ROD 28. 5

RODS FOR WHICH NO SCRAM TIME IS AVAILABLE STATUS 0

ROD	24.61	28.61	36.61	44.61											
ROD	12.57	16.57	20.57	24.57	28.57	32.57	36.57	44.57	48.57	52.57					
ROD	8.53	12.53	16.53	20.53	24.53	28.53	32.53	36.53	40.53	44.53	48.53	52.53	56.53		
ROD	12.49	16.49	20.49	24.49	28.49	32.49	36.49	40.49	44.49	48.49	52.49	56.49			
ROD	4.45	8.45	12.45	16.45	20.45	24.45	28.45	32.45	36.45	40.45	44.45	48.45	52.45	56.45	60.45
ROD	4.41	8.41	12.41	16.41	20.41	24.41	28.41	32.41	36.41	40.41	44.41	48.41	52.41	56.41	60.41
ROD	4.37	8.37	12.37	16.37	20.37	24.37	28.37	32.37	36.37	40.37	44.37	48.37	52.37	56.37	
ROD	4.33	8.33	12.33	16.33	20.33	24.33	28.33	32.33	36.33	40.33	44.33	48.33	52.33	56.33	60.33
ROD	4.29	8.29	12.29	16.29	20.29	24.29	28.29	32.29	36.29	40.29	44.29	48.29	52.29	56.29	60.29
ROD	4.25	8.25	12.25	16.25	20.25	24.25	28.25	32.25	36.25	40.25	44.25	48.25	52.25	56.25	60.25
ROD	4.21	8.21	12.21	16.21	20.21	24.21	28.21	32.21	36.21	40.21	44.21	48.21	52.21	56.21	60.21
ROD	8.17	12.17	16.17	20.17	24.17	28.17	32.17	36.17	40.17	44.17	48.17	52.17	56.17		
ROD	8.13	12.13	16.13	20.13	24.13	28.13	32.13	36.13	40.13	44.13	48.13	52.13	56.13		
ROD	12. 9	16. 9	20. 9	24. 9	28. 9	32. 9	36. 9	40. 9	44. 9	48. 9	52. 9				
ROD	20. 5	24. 5	32. 5	36. 5	40. 5	44. 5									

4.2.8 Sample Output - Timing Analysis

CONTROL TIMING ANALYSIS PERFORMED AT 12:25 AM THU , 28 JUNE, 1984

RUN NUMBER 38 SCRAM TIME TEST ROD 48-39

CRD I.D.							STATUS	OF	ALL	CRD'S					
J															
61					1	Ø	Ø	3	Ø	2	Ø				
57			Ø	Ø	Ø	Ø	Ø	Ø	Ø	4	Ø	Ø	Ø		
53		Ø	Ø	3	Ø	Ø	Ø	Ø	Ø	Ø	6	Ø	Ø	Ø	
49		99	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	
45	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
41	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
37	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	99
33	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
29	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
25	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
21	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø
17		Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	
13		Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	
9			Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø	Ø		
5					Ø	Ø	99	Ø	Ø	Ø	Ø				
1	4	8	12	16	2Ø	24	28	32	36	4Ø	44	48	52	56	6Ø

Ø => NO DATA, 1 => FAST ROD, 2 => NORMAL ROD, 3 => SLOW ROD, 4 => FAILED ROD, 99 => INOPERATIVE ROD

4.3.2 DSPLY PRINT Sample Output

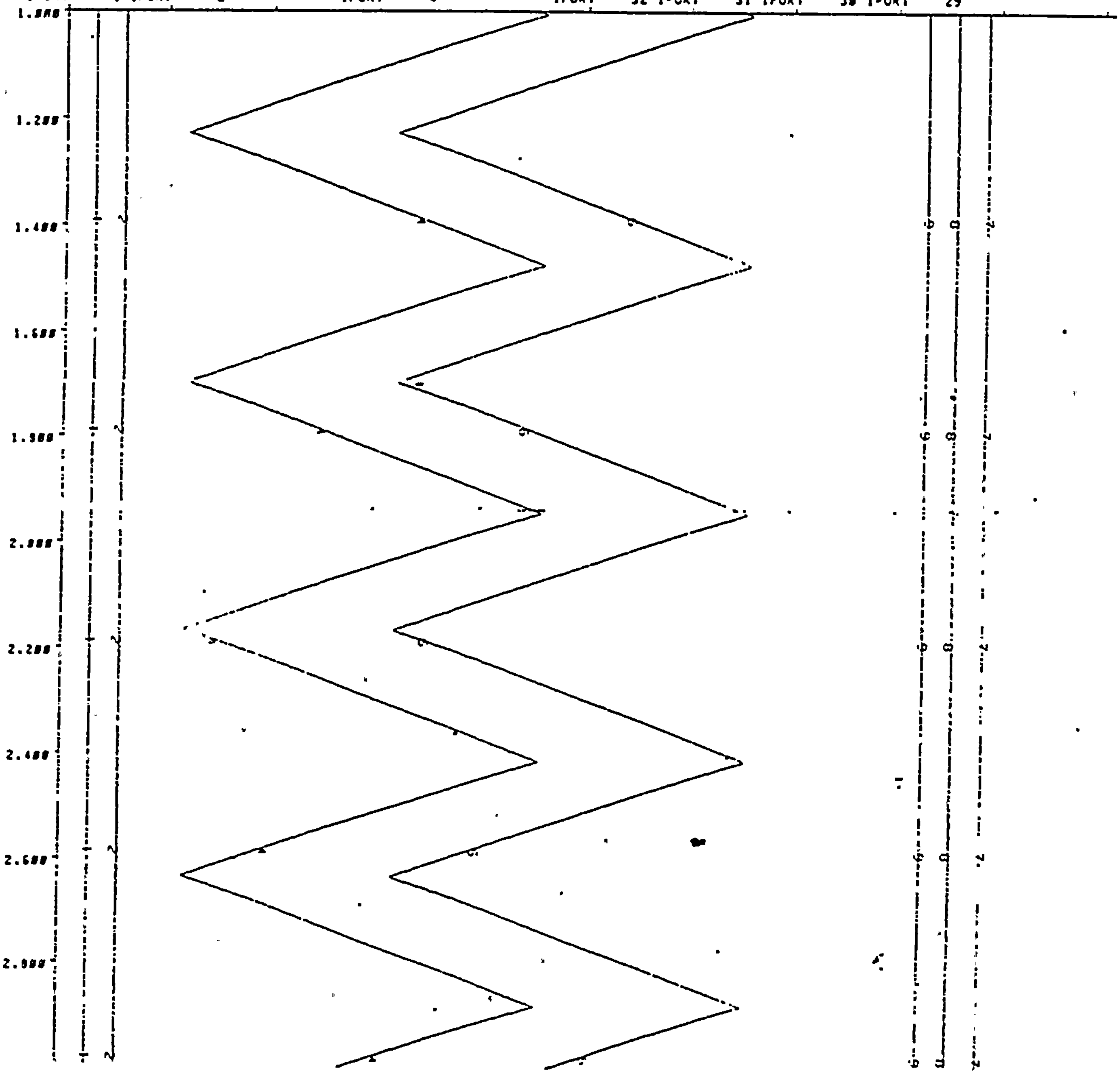
```
                    INZ-1
                    FILE #    1
                    RUN  #    1
                    DATE    1/ 1/85
                    TIME @ START OF RUN    4:19:21:313
                    BEGINNING AT    1.00 SECS. FROM START OF RUN

CHAN#      97 CHAN#       98 CHAN#       99 CHAN#      100
CHA  097      CHA  098       CHA  099       CHA  100
MV            MV             MV             MV
LINK       11 LINK        11 LINK        11 LINK.      11
ISUB.       1 ISUBL        1 ISUB.        1 ISUBL       1
IPORT       1 IPORT        2 IPORT        3 IPORT       4
RELATIVE TIME . . . 0            0            0              2
        2865.600     5764.800     8716.801     6086.400
        2860.800     5760.000     8712.000     6187.200
        2860.800     5764.800     8716.801     6297.601
        2860.800     5764.800     8716.801     6398.400
        2860.800     5760.000     8716.801     6326.400
        2860.800     5760.000     8716.801     6139.200
        2860.600     5764.800     8712.000     6009.601
        2860.800     5764.800     8716.801     5870.400
        2860.800     5760.000     8716.801     5764.800
        2865.600     5760.000     8716.801     5616.000
RELATIVE TIME . . . 0            0            0             22
        2860.800     5764.800     8716.801     5438.400
        2860.800     5764.800     8712.000     5328.000
        2860.800     5764.800     8712.000     5203.200
        2860.800     5760.000     8712.000     5070.400
        2860.800     5764.800     8712.000     4944.000
        2860.800     5760.000     8712.000     4790.400
        2860.800     5760.000     8716.801     4651.200
        2860.800     5764.800     8712.000     4540.800
        2865.600     5760.000     8716.801     4416.000
        2860.800     5764.800     8716.801     4305.601
RELATIVE TIME . . . 0            0            0             42
        2865.600     5760.000     8716.801     4147.200
        2860.800     5760.000     8716.801     4012.800
        2860.800     5760.000     8716.801     3897.600
        2865.600     5760.000     8716.801     3748.800
        2860.800     5764.800     8716.801     3652.800
        2865.600     5760.000     8716.801     3513.600
        2865.600     5760.000     8716.801     3398.400
        2860.800     5764.800     8716.801     3278.400
        2865.600     5760.000     8716.801     3139.200
        2860.800     5760.000     8716.801     3033.600
RELATIVE TIME . . . 0            0            0             62
        2865.600     5760.000     8712.000     2875.200
        2860.800     5764.800     8716.801     2740.800
        2860.800     5764.800     8716.801     2649.600
        2860.800     5764.800     8716.801     2520.000
        2865.600     5760.000     8716.801     2409.600
        2860.800     5764.800     8716.801     2284.800
        2865.600     5764.800     8716.801     2126.400
        2860.800     5764.800     8716.801     2054.400
        2865.600     5764.800     8716.801     1924.800
        2865.600     5760.000     8712.000     1780.800
RELATIVE TIME . . . 0            0            0             82
        2860.800     5764.800     8716.801     1665.600
        2865.600     5760.000     8712.000     1531.200
        2865.600     5764.800     8716.801     1440.000
        2865.600     5764.800     8721.600     1310.400
        2865.600     5764.800     8716.801     1195.200
        2860.800     5764.800     8716.801     1084.800
        2860.800     5764.800     8716.801      940.800
        2865.600     5760.000     8716.801      811.200
        2865.600     5760.000     8716.801      676.800
        2860.800     5760.000     8712.000      580.800
```

4.3.4 DSPLY PLOT Sample Output

O F F - L I N E   P L O T
INZ-1
FILE #   1
RUN #   1
DATE   1/ 1/85
TIME @ START OF RUN   4:19:21:313
BEGINNING AT   1.00 SECS. FROM START OF RUN

.200SEC./GRID LINE

CHAN#	97	98	100	96	95	94	93
CHA	097	098	100	096	095	094	093
MV							
EU	10000.0	10000.0	4000.0	4000.0	10000.0	10000.0	10000.0
OFF	0.0	12000.0	0.0	0.0	-20000.0	-10000.0	0.0
ORG.	1.	2.	4.	6.	7.	8.	9.
LINK	11	11	11	11	11	11	11
ISUBL	1	1	1	1	1	1	1
IPORT	1	2	4	32	31	30	29

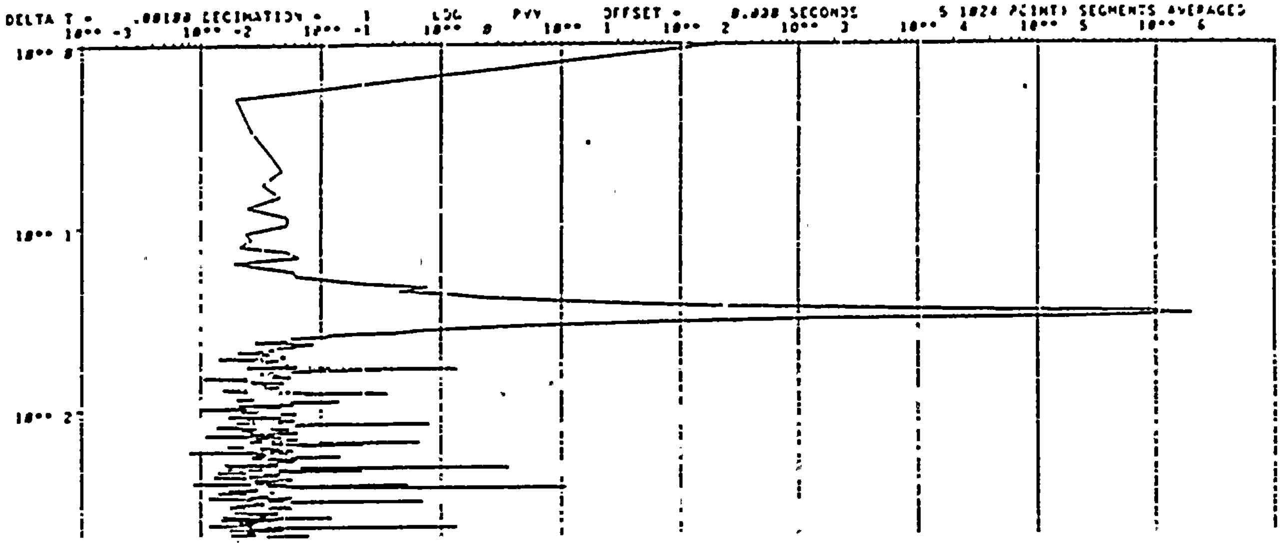


ATTACHMENT 10

DYNNO Sample Output Con't

RUN 1 DATE 12/ 2.83 TIME 12:18:58. 8 SINE WAVE 30HZ

DELTA T = .00100 DECIMATION = 1 LOG PVV OFFSET = 0.000 SECONDS 5 1024 POINT SEGMENTS AVERAGED



X CHAN 1 CHAN 001 G MEAN 0.192 STD DE.264.7595 Y CHAN 1 CHAN 261 G MEAN 16.765 STD DEV1775.056
X LP FILTER= 10.137 E 3 POLE Y LP FILTER= 0.220 E 0 POLE
SPECTRUM VALUES
FREQUENCIES 0.00 TO 25.00 25.00 TO 50.00 50.00 TO125.00 125 00 TO250.00 250.00 TO500.00 0.00 TO500.00
RMS VALUES .7901E+01 ..778E+04 .2693E+01 5186E+01 .6432E+01 .1778E+04

4.10.2 HISTOGRAM Sample Output

```
:RU,HIST
 ENTER COMMENTS                                             HIST-2
 FILE NAME?   (ENTER -1 FOR MILLIVOLT DISPLAY)              MILLV
 ENTER A  1  FOR CRT DISPLAY                                0
 1ST CHANNEL,  LAST CHANNEL,  # OF SCANS?                   21,31,500
 ENTER -1 TO END                                            -1
```

HISTOGRAM   4:14 AM TUE., 1 JAN., 1985

HIST-2

ENG. UNITS DISPLAY   FILE = MILLV

CHAN	NAME	UNITS	MEAN	MAX	MIN	SIG	P-P	LO5%	LO1%	LO.2%	NORM	HI.2%	HI1%	HI5%
21	CHA 021	MV	2883.0435	2889.6001	2875.2002	2.3903	14.3999	0	0	1	498	1	0	0
22	CHA 022	MV	5768.7000	5769.6006	5764.7998	2.0566	4.8008	0	0	0	500	0	0	0
23	CHA 023	MV	8724.2402	8726.4004	8716.8008	2.9725	9.5996	0	0	0	500	0	0	0
24	CHA 024	MV	-859.7472	6369.6006	-7185.6006	3824.8179	13555.2010	258	2	0	0	1	0	239
25	CHA 025	MV	2875.2769	2880.0000	2870.3999	1.7637	9.6001	0	0	0	500	0	0	0
26	CHA 026	MV	5765.5391	5769.6006	5760.0000	3.3612	9.6006	0	0	0	500	0	0	0
27	CHA 027	MV	8724.5664	8726.4004	8712.0000	2.9668	14.4004	0	0	0	500	0	0	0
28	CHA 028	MV	-859.5168	6379.2002	-7228.7998	3824.7651	13608.0000	258	2	0	0	0	2	238
29	CHA 029	MV	2877.3696	2880.0000	2870.3999	2.4116	9.6001	0	0	2	498	0	0	0
30	CHA 030	MV	5763.0332	5764.7998	5755.2002	2.2659	9.5996	0	0	0	500	0	0	0
31	CHA 031	MV	8717.7227	8721.5996	8712.0000	2.3900	9.5996	0	0	0	500	0	0	0

ATTACHMENT 12

4.13.2 MPXT Sample Output

```
:RU,MPXT
 ENTER COMMENTS                                              MPXT-2
 FILE NAME?   (ENTER -1 FOR MILLIVOLT DISPLAY)               MILLV
 ENTER A  1  FOR CPT DISPLAY.                                0
 1ST CHANNEL, LAST CHANNEL, # OF SCANS?                      246,256,10
 ENTER A -1 TO END.                                          -1
  MPXT  STOP  0000
```

MPXT   COMMENTS   MPXT-2
-- ENG. UNITS DISPLAY - FILE = MILLV

POINT ID. =	CHA 246	CHA 247	CHA 248	CHA 249	CHA 250	CHA 251	CHA 252	CHA 253	CHA 254	CHA 255	CHA 256
ENG. UNITS=	MV	MV	MV	MV	MV	MV	MV	MV	MV	MV	MV
LINK =	11	11	11	11	11	11	11	11	11	11	11
SUBLINK =	1	1	1	1	1	1	1	1	1	1	1
PORT =	22	23	24	25	26	27	28	29	30	31	32
SLOPE =	1.200	1.200	1.200	1.200	1.200	1.200	1.200	1.200	1.200	1.200	1.200
INTERCEPT =	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
	5769.601	8726.400	619.200	2875.200	5764.800	8721.600	684.000	2975.200	5764.800	8721.600	595.200
	5764.800	8726.400	-1525.400	2975.200	5764.800	8721.600	-1562.000	2902.000	5764.000	8716.321	-1525.200
	5769.601	8726.400	-3523.600	2870.400	5769.601	8721.600	-3613.200	2875.200	5764.800	8716.901	-3229.600
	5764.800	8721.600	-5563.000	2975.200	5769.601	8726.400	-5601.601	2580.000	5760.000	8716.321	-5565.000
	5764.800	8716.801	-6724.800	2575.200	5764.800	8726.400	-6744.000	2875.200	5764.800	8721.600	-6777.601
	5769.601	8726.400	-4724.000	2575.200	5769.601	8721.600	-4694.900	2975.200	5764.800	8716.321	-4723.200
	5769.601	8721.600	-2651.200	2575.200	5769.601	8726.400	-2823.200	2380.000	5760.000	8712.000	[illegible]
	5769.601	8726.400	-4339.200	2575.200	5764.800	8726.400	-4392.400	2350.000	5764.800	8716.321	[illegible]
	5769.601	8726.400	-6335.000	2975.200	5764.800	8721.600	-6394.000	2360.000	5760.000	8716.301	-6379.200
	5769.601	8721.600	-6223.621	2975.200	5764.800	8721.600	-6883.601	2975.200	5764.800	8721.600	-6229.621

ATTACHMENT 13

4.28.2 VMEAN Sample Output

RUN NO. 1 DATE 1/ 1/85 TIME 4:19:22:315 THRU 4:19:24:315

RUN DESCRIPTION : INZ-1

SCAN PERIOD (MS) = 2.00

FIRST CHANNEL = 1

LAST CHANNEL = 100

I. D. T A B L E

						----------------ABSOLUTE	VALUES----------------			
CH#	NAME		EU	LINK	SUBL	PORT	MEAN	STD DEV	MIN VAL	MAX VALUE	P-P
1	CHA	001	MV	11	1	1	2862.910	2.412	2856.000	2865.600	9.600
2	CHA	002	MV	11	1	2	5762.561	2.335	5755.200	5764.800	9.600
3	CHA	003	MV	11	1	3	8715.856	1.191	8712.000	8721.600	9.600
4	CHA	004	MV	11	1	4	-286.860	3896.320	-7228.800	6417.601	13646.400
5	CHA	005	MV	11	1	5	2874.064	2.034	2870.400	2875.200	4.800
6	CHA	006	MV	11	1	6	5757.583	2.662	5750.400	5760.000	9.600
7	CHA	007	MV	11	1	7	8718.757	2.797	8712.000	8726.400	14.400
8	CHA	008	MV	11	1	8	-282.736	3896.001	-7219.200	6427.200	13646.400
9	CHA	009	MV	11	1	9	2873.747	2.208	2870.400	2875.200	4.800
10	CHA	010	MV	11	1	10	5762.623	2.308	5755.200	5764.800	9.600
11	CHA	011	MV	11	1	11	8718.120	2.362	8712.000	8721.600	9.600
12	CHA	012	MV	11	1	12	-286.549	3896.175	-7238.400	6403.200	13641.602
13	CHA	013	MV	11	1	13	2871.493	2.134	2865.600	2875.200	9.600
14	CHA	014	MV	11	1	14	5759.779	.976	5755.200	5760.000	4.800
15	CHA	015	MV	11	1	15	8715.621	1.262	8712.000	8721.500	9.500
16	CHA	016	MV	11	1	16	-286.462	3895.834	-7238.400	6451.200	13689.602
17	CHA	017	MV	11	1	17	2874.064	2.034	2870.400	2875.200	4.800
18	CHA	018	MV	11	1	18	5758.624	2.005	5755.200	5760.000	4.800
19	CHA	019	MV	11	1	19	8723.955	3.272	8712.000	8726.400	14.400
20	CHA	020	MV	11	1	20	-280.646	3896.066	-7224.000	6403.200	13627.201
21	CHA	021	MV	11	1	21	2883.295	2.300	2875.200	2889.600	14.400
22	CHA	022	MV	11	1	22	5768.675	2.076	5764.800	5769.601	4.801
23	CHA	023	MV	11	1	23	8724.180	3.050	8712.000	8726.400	14.400
24	CHA	024	MV	11	1	24	-287.666	3896.891	-7228.800	6403.200	13632.000
25	CHA	025	MV	11	1	25	2875.378	1.607	2870.400	2880.000	9.600
26	CHA	026	MV	11	1	26	5765.351	2.948	5760.000	5769.601	9.601
27	CHA	027	MV	11	1	27	8724.592	2.807	8716.801	8726.400	9.602
28	CHA	028	MV	11	1	28	-287.431	3895.583	-7228.800	6417.601	13646.400
29	CHA	029	MV	11	1	29	2877.238	2.383	2870.400	2880.000	9.600
30	CHA	030	MV	11	1	30	5762.978	2.233	5755.200	5764.800	9.600
31	CHA	031	MV	11	1	31	8717.500	2.520	8712.000	8721.600	9.600
32	CHA	032	MV	11	1	32	-284.100	3896.834	-7228.800	6412.800	13641.600
33	CHA	033	MV	11	1	1	2862.896	2.401	2856.000	2865.600	9.600
34	CHA	034	MV	11	1	2	5762.517	2.329	5755.200	5764.800	9.600
35	CHA	035	MV	11	1	3	8715.832	1.173	8712.000	8721.600	9.600
36	CHA	036	MV	11	1	4	-286.501	3896.007	-7228.800	6417.601	13646.400
37	CHA	037	MV	11	1	5	2874.083	2.022	2870.400	2875.200	4.800
38	CHA	038	MV	11	1	6	5757.574	2.670	5750.400	5760.000	9.600
39	CHA	039	MV	11	1	7	8718.838	2.775	8712.000	8726.400	14.400
40	CHA	040	MV	11	1	8	-283.772	3896.101	-7219.200	6427.200	13646.400
41	CHA	041	MV	11	1	9	2873.762	2.202	2870.400	2875.200	4.800
42	CHA	042	MV	11	1	10	5762.604	2.320	5755.200	5764.800	9.600
43	CHA	043	MV	11	1	11	8718.095	2.467	8712.000	8721.600	9.600
44	CHA	044	MV	11	1	12	-286.673	3895.914	-7238.400	6403.200	13641.602
45	CHA	045	MV	11	1	13	2871.536	2.159	2865.600	2875.200	9.600
46	CHA	046	MV	11	1	14	5759.779	.976	5755.200	5760.000	4.800
47	CHA	047	MV	11	1	15	8715.650	1.256	8712.000	8721.600	9.600
48	CHA	048	MV	11	1	16	-286.553	3896.155	-7238.400	6451.200	13689.602
49	CHA	049	MV	11	1	17	2873.977	2.008	2870.400	2875.200	4.800

<u>Section 1.2 C</u>

Generic Letter 83-28

Post-Trip Review - Data and Information Capability

Safety Parameter Display System (SPDS)

2. Capability for assessing the time history of analog variables needed to determine the cause of unscheduled reactor shutdowns, and the functioning of safety-related equipment.

1.2.2.1 Brief description of equipment (e.g., plant computer, dedicated computer, strip charts).

NMP2 Response

The Nine Mile Point 2 Liquid Radwaste System and Emergency Response Facility computer system (LWCS/ERF) consists of dual Honeywell 4500 CPU's on which the standard Honeywell SEER software package has been implemented and modified as needed. Each processor contains 25K of core memory and dual ported Ampex large core stores used for bulk devices. The system also utilizes two 80MB disk drives for additional storage capacity. A magnetic tape unit is used for historical recording and additional back-up capabilities. Video monitors, types/printers and keyboards are located in the computer room, the control room, the Technical Support Center, and the Emergency Operations Facility to enable operators in recording and viewing event data.

1.2.2.2 Parameters monitored, sampling rate, and basis for selecting parameters and sampling rate.

NMP2 Response

There can be up to 690 data points contained on the Event Historical Recording System. These points can be contained in one of 115 groups made up of 6 data points each. Depending on which group the data point is contained in. The sampling rate can be every 1, 5, or 30 seconds. The only exception to these rates is with the two week post event recording. This data is collected every 15 minutes.

1.2.2.3 Duration of time history (minutes before trip and minutes after trip).

NMP2 Response

There are three types of historical event recording on the LWCS/ERF computer system. Two hours of pre-event data is collected on a continuous basis in a circular buffer. When a pre-defined event is detected, the buffer is frozen. Twelve hours of post-event data is collected immediately following the occurrence of a predefined event. This is done by the use of two 1 hour buffers which are used in a switching process for 12 hours. Two weeks of additional data will be collected immediately after the 12 hour collection is completed.

1.2.2.4 Format for displaying data including scale (readability) of time histories.

NMP2 Response

The Trend history consists of five secondary displays which are reactivity control, core cooling, coolant system integrity, and containment integrity (Attachments A-D) Radioactive Release (Future). Each secondary display consists of a number of trend plots covering a 6-minute time span. The reactivity control display consists of trend plots of APRM power, IRM power, and SRM log count rate. The core cooling display consists of a trend plot of RPV water level. The coolant system integrity display consists of trend plots of RPV pressure and drywell pressure. The containment integrity display consists of trend plots for drywell pressure, drywell oxygen concentration, suppression pool temperature, and suppression pool water level, as well as the containment isolation valve groups. The radioactive release display is a composite of Stack, Offgas and Containment Rad Monitor Parameters.

All displays contain safety function blocks at the bottom, which may be green or red depending upon whether the function is considered "normal" or "in alarm/unknown." The color of the safety functions is determined by the status of the variables associated with those safety functions.

Attachments (E-H) are the formats used for viewing the Event Historical data. This data can be used by displaying it on a video or printing it to a typer. The operator can view the data based on time for each sample taken. This display/printout can be based on any of the groups and ranged over all or any of the time period of the event recording. The operator can also display a trend of the various groups. This trend can also be based by group and consist of data over a specified period of the data recording.

1.2.2.5 Capability for retention of data, information, and physical evidence (both hardware and software).

NMP2 Response

Primary retention of data is done on disk buffers for all three types of historical recording. Two buffers are used for the pre-event recording, which is able to hold 1 hour's worth of data. Upon the detection of an event, the pre-event buffers will be frozen and can be saved to magnetic tape by a programmer. The 12 hour post-event data is collected with the use of two, 1 hour buffers on disk. These buffers are used in a switching mode, and are dumped to magnetic tape when they become full. The 14 day post event collection also uses two buffers on the disk. These buffers are dumped to magnetic tape daily, by the programmer. The programmer has the option of dumping the data to tape manually or initiating an automatic mode. The data can be restored from tape to a review buffer on the disk to allow a programmer to view or trend the data.

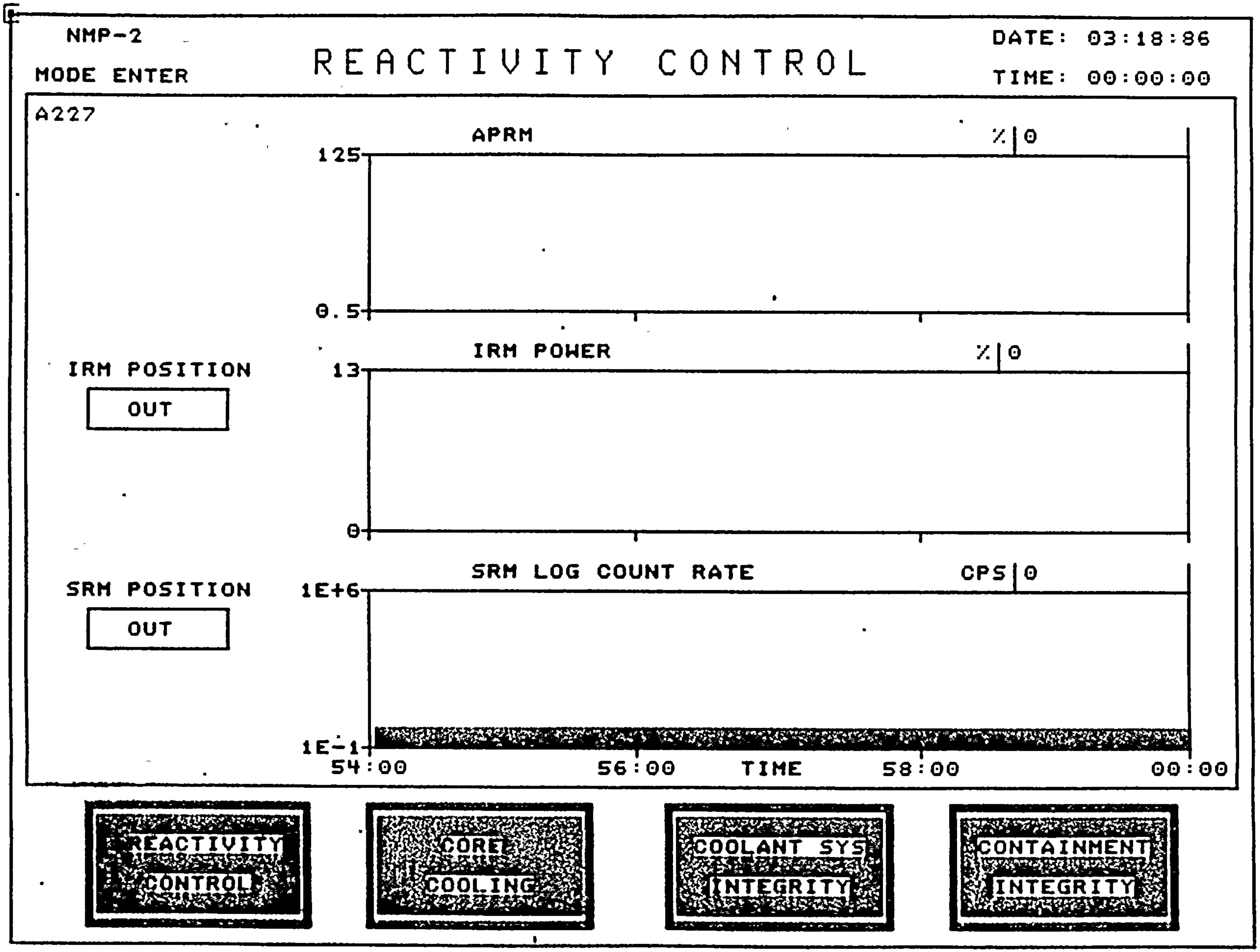
1.2.2.6 Power source(s) (e.g., Class IE, non-Class IE, non-interruptable).

NMP2 Response

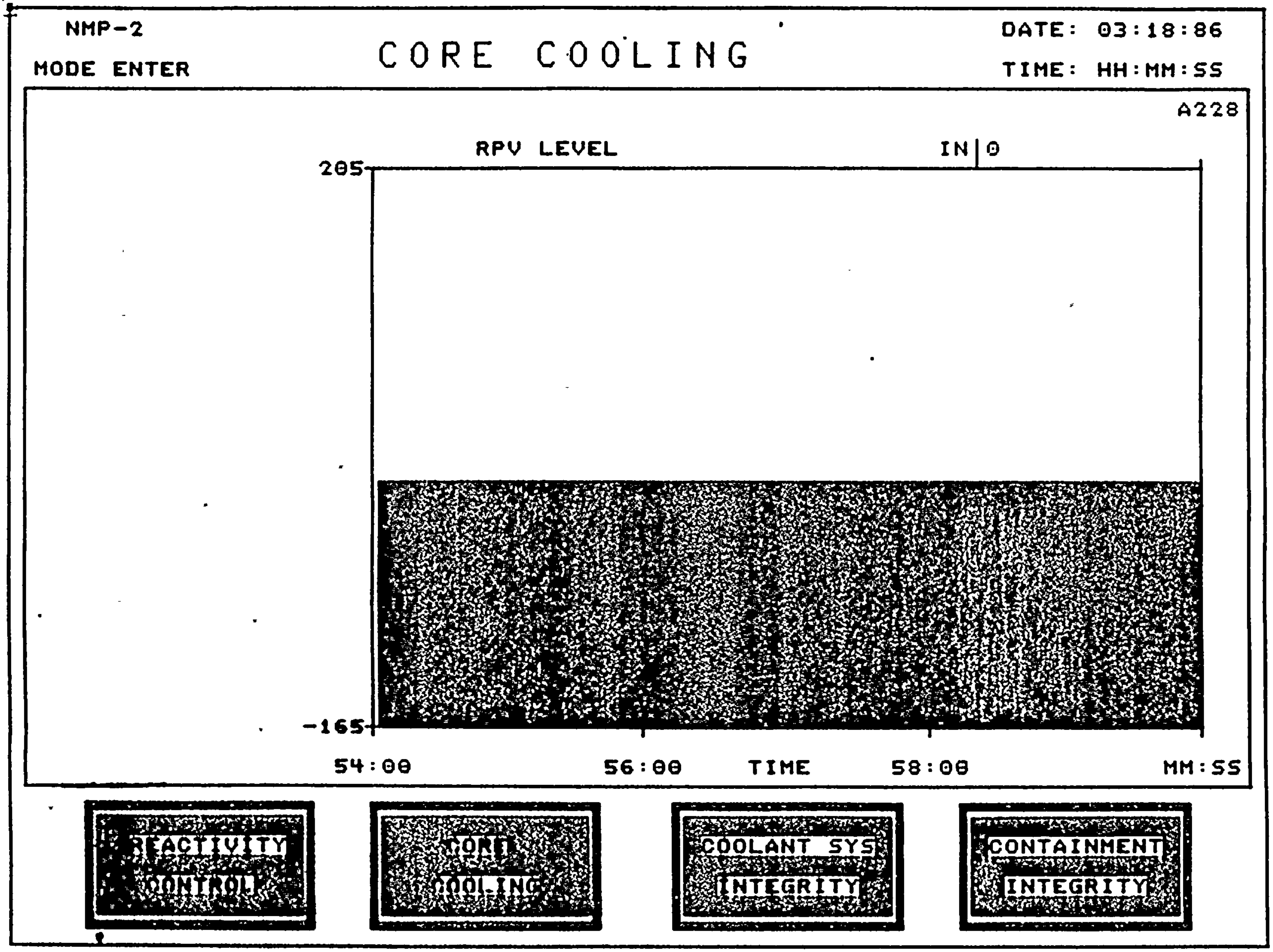
Power to the Unit 2 LWCS/ERF computer is provided by an uninterruptible power supply 2VBB-UPS1A non class 1E. This supply is fed from a 600V power panel 2VBB-PNL301, which is supplied by either the station generator 13.8kv line (2WJS-US3, during normal operation) or from an off-site Scriba 115kv line (2NJS-US4, during a shutdown condition). Back-up power is supplied by a 125v DC battery supply, 2BYS-SWG001A.

In summary, upon loss of normal power, a static transfer switch transfers power from the normal source to the alternative source. If both normal and alternate sources are lost, the DC source will automatically pickup the loads by means of a DC auctioneering circuit.

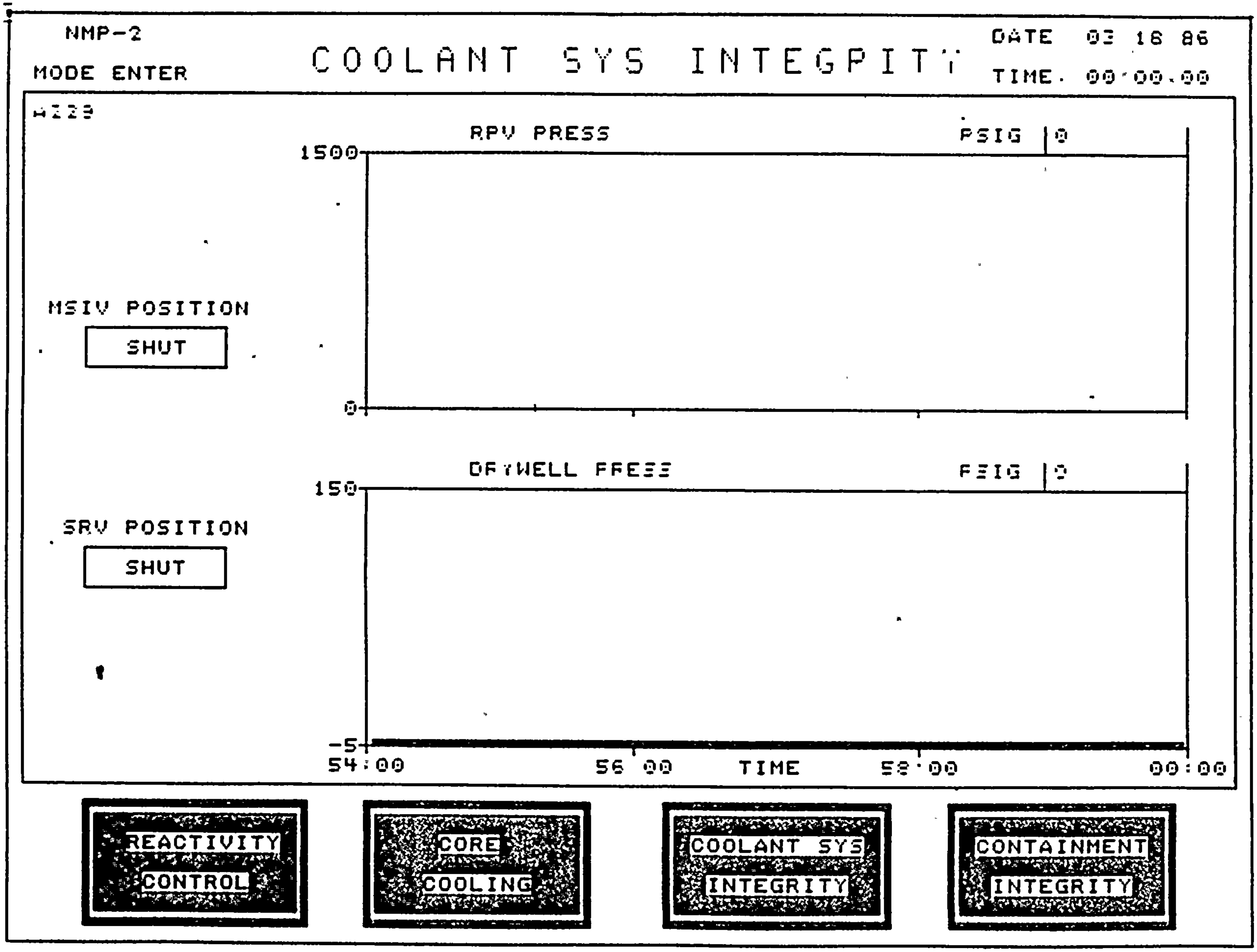
ATTACHMENT A



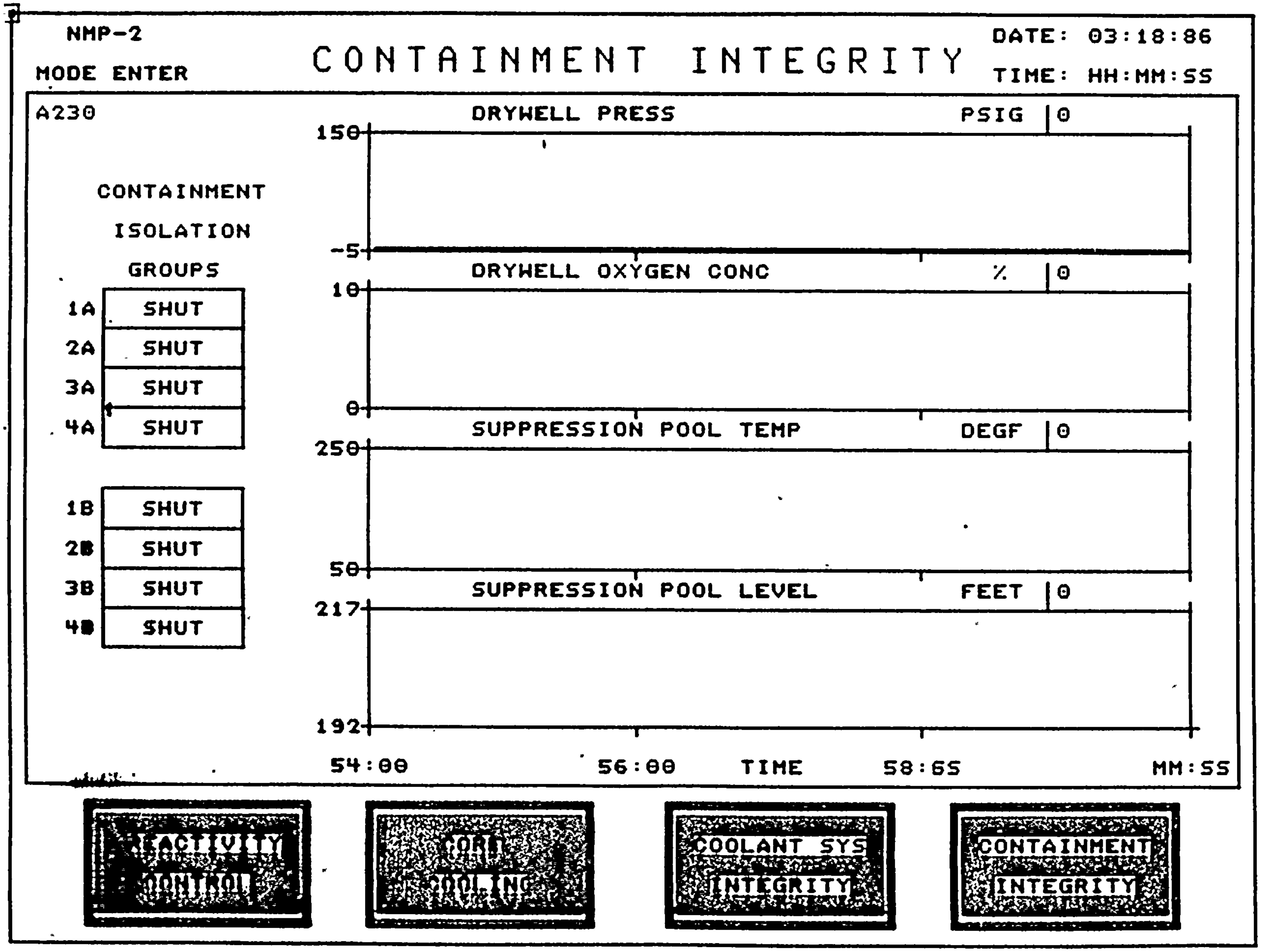
ATTACHMENT B



ATTACHMENT C



ATTACHMENT D

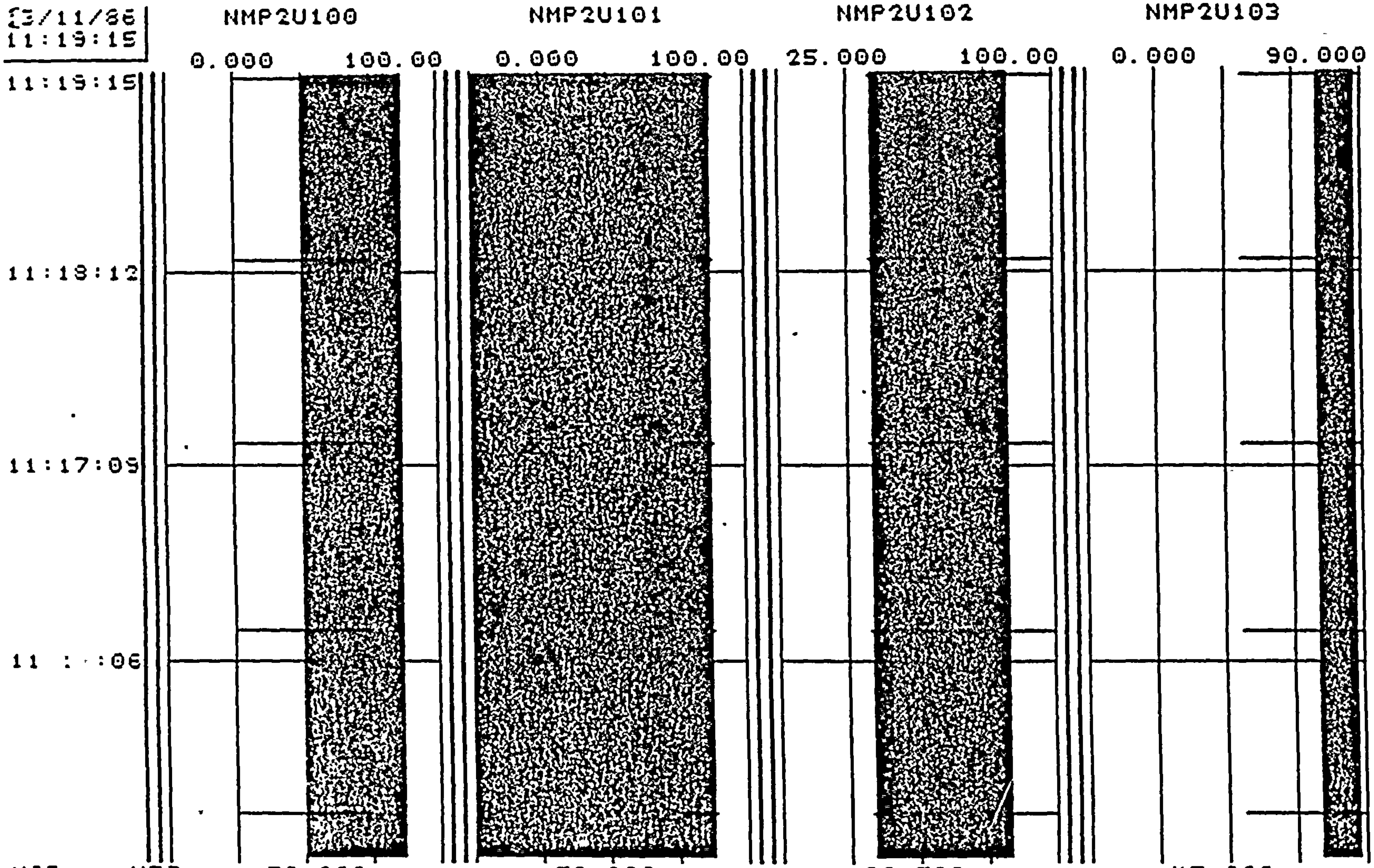


ATTACHMENT E

ENTER (D) DISPLAY, (PF) PAGE FORWARD, (PB) PAGE BACK, (P) PRINT, (C) CANCEL PRNT
REVIEW BUFFER TIME SPAN 03-11-86 11:14 THRU 03-12-86 18:51
START DATE 03-11-86 TIME 11:14, END DATE 03-11-86 TIME 20:11. GROUP NUMBER 1

TIME	NMP2U100 %PWR	NMP2U101 %PWR	NMP2U102 %PWR	NMP2U103 %PWR	NMP2U104 %PWR	NMP2U105 %PWR
3-11-86						
11:14:06	86.19	86.75	86.97	87.13	86.09	87.40
11:14:07	86.19	86.75	86.97	87.13	86.09	87.40
11:14:08	86.19	86.75	86.97	87.13	86.09	87.40
11:14:09	86.19	86.75	86.97	87.13	86.09	87.40
11:14:10	86.19	86.75	86.97	87.13	86.09	87.40
11:14:11	86.19	86.75	86.97	87.13	86.09	87.40
11:14:12	86.19	86.75	86.97	87.13	86.09	87.40
11:14:13	86.19	86.75	86.97	87.13	86.09	87.40
11:14:14	86.19	86.75	86.97	87.13	86.09	87.40
11:14:15	86.19	86.75	86.97	87.13	86.09	87.40
11:14:16	86.19	86.75	86.97	87.13	86.09	87.40
11:14:17	86.19	86.75	86.97	87.13	86.09	87.40
11:14:18	86.19	86.75	86.97	87.13	86.09	87.40
11:14:19	86.19	86.75	86.97	87.13	86.09	87.40
11:14:20	86.19	86.75	86.97	87.13	86.09	87.40
11:14:21	86.19	86.75	86.97	87.13	86.09	87.40
11:14:22	86.19	86.75	86.97	87.13	86.09	87.40
11:14:23	86.19	86.75	86.97	87.13	86.09	87.40
11:14:24	86.19	86.75	86.97	87.13	86.09	87.40
11:14:25	86.19	86.75	86.97	87.13	86.09	87.40
11:14:26	86.19	86.75	86.97	87.13	86.09	87.40
11:14:27	86.19	86.75	86.97	87.13	86.09	87.40
11:14:28	86.19	86.75	86.97	87.13	86.09	87.40
11:14:29	86.19	86.75	86.97	87.13	86.09	87.40
11:14:30	86.19	86.75	86.97	87.13	86.09	87.40
11:14:31	86.19	86.75	86.97	87.13	86.09	87.40
11:14:32	86.19	86.75	86.97	87.13	86.09	87.40
11:14:33	86.19	86.75	86.97	87.13	86.09	87.40
11:14:34	86.19	86.75	86.97	87.13	86.09	87.40
11:14:35	86.19	86.75	86.97	87.13	86.09	87.40
11:14:36	86.19	86.75	86.97	87.13	86.09	87.40
11:14:37	86.19	86.75	86.97	87.13	86.09	87.40
11:14:38	86.19	86.75	86.97	87.13	86.09	87.40
11:14:39	86.19	86.75	86.97	87.13	86.09	87.40
11:14:40	86.19	86.75	86.97	87.13	86.09	87.40
11:14:41	86.19	86.75	86.97	87.13	86.09	87.40
11:14:42	86.19	86.75	86.97	87.13	86.09	87.40
11:14:43	86.19	86.75	86.97	87.13	86.09	87.40
11:14:44	86.19	86.75	86.97	87.13	86.09	87.40
11:14:45	86.19	86.75	86.97	87.13	86.09	87.40

ATTACH[illegible]T F

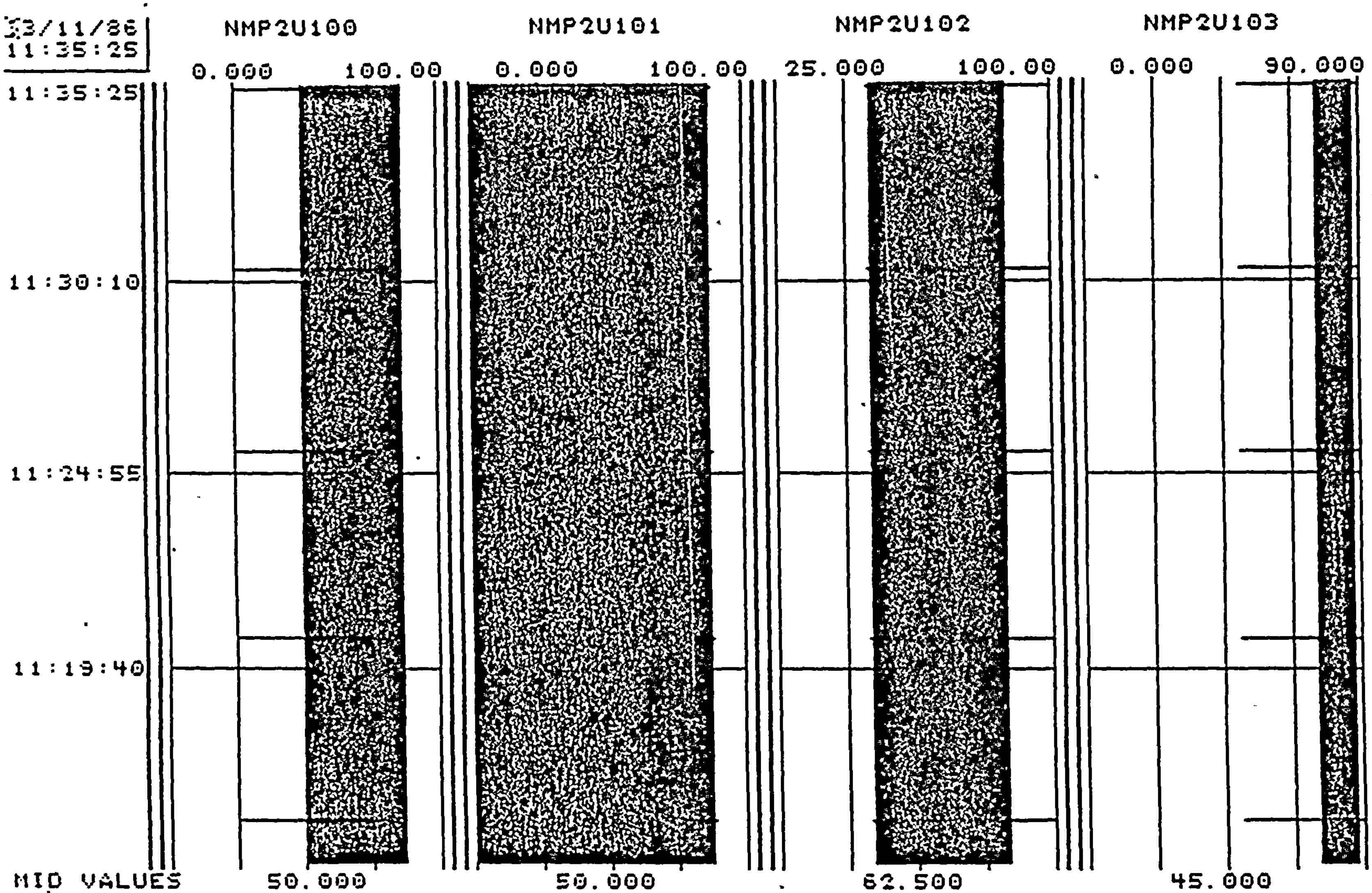


MID .UES 50.000 50.000 62.500 45.000

GROL)21 INTERVAL 01 SEC — TREND LIMITS —

		VALUE	LOW	HIGH
NMF[illegible] 0	APRM CHANNEL A PWR LVL	85.78	25.00	75.000
NMF[illegible] 1	APRM CHANNEL B PWR LVL	86.94	75.00	90.000
NMF[illegible] 2	APRM CHANNEL C PWR LVL	86.81	50.00	100.00
NMF[illegible] 3	APRM CHANNEL D PWR LVL	87.15	50.00	90.000

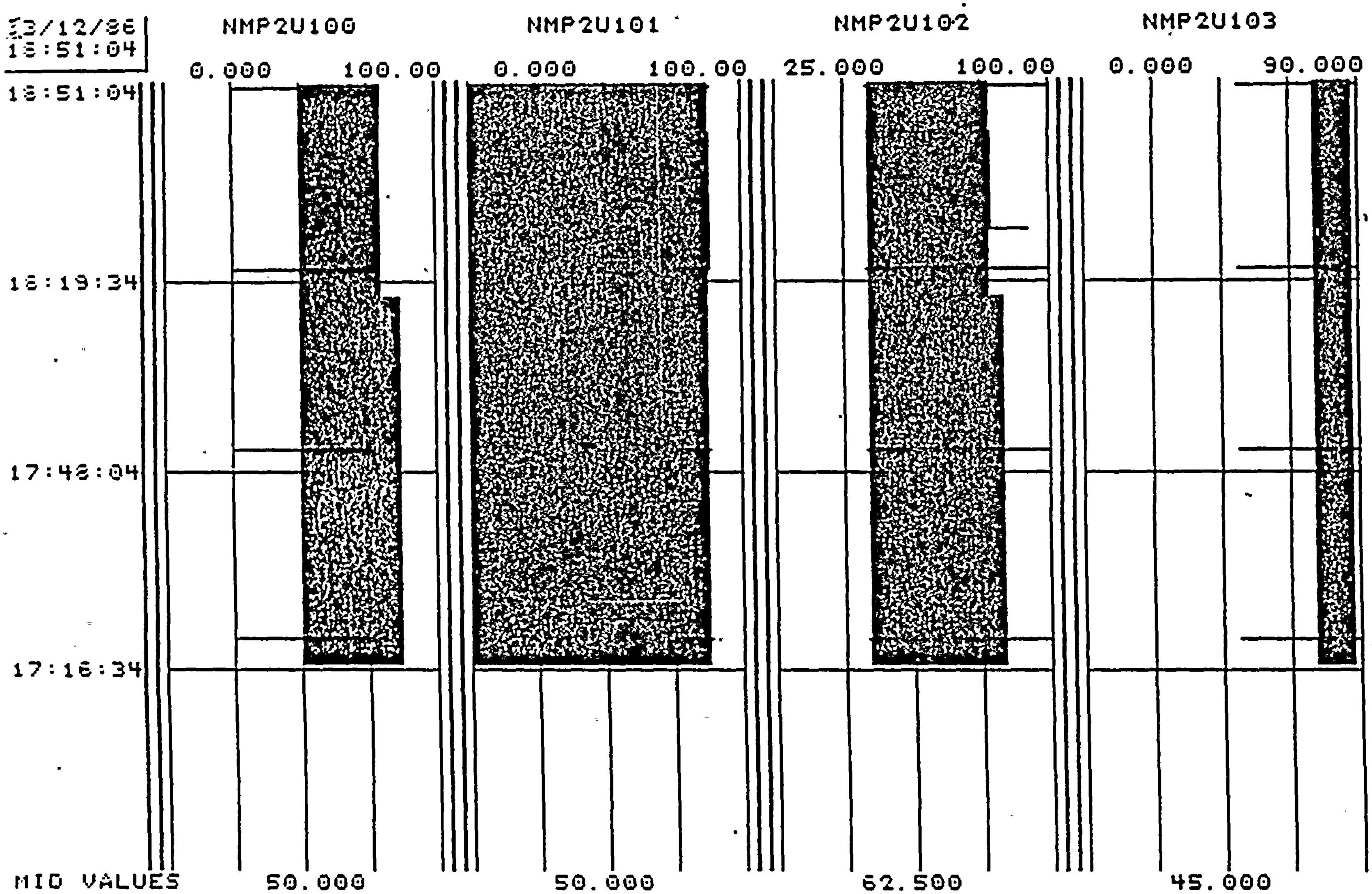
ATTACHME



GROUP 021 INTERVAL 05 SEC

		VALUE	- TREND LIMITS - LOW	HIGH
NMP2U100	APRM CHANNEL A PWR LVL	85.78	25.00	75.000
NMP2U101	APRM CHANNEL B PWR LVL	86.94	75.00	90.000
NMP2U102	APRM CHANNEL C PWR LVL	86.81	50.00	100.00
NMP2U103	APRM CHANNEL D PWR LVL	87.15	50.00	90.000

ATTACHME



GROUP 021 INTERVAL 30 SEC

		VALUE	- TREND LIMITS - LOW	HIGH
NMP2U100	APRM CHANNEL A PWR LVL	85.78	25.00	75.000
NMP2U101	APRM CHANNEL B PWR LVL	86.94	75.00	90.000
NMP2U102	APRM CHANNEL C PWR LVL	86.81	50.00	100.00
NMP2U103	APRM CHANNEL D PWR LVL	87.15	50.00	90.000

<u>Section 2.1, 2.2</u>

Generic Letter 83-28

Equipment Classification and Vendor Interface

2.1 (Cont'd)

However, a task is currently underway to upgrade the details of our equipment classification list (Q-List, See Response 2.2.1.2). This will provide additional assurance that those components which contribute to the reactor trip function are appropriately classified as safety-related.

Administrative controls consisting of documents, procedures and information handling systems are used in the station to control safety-related activities including maintenance, work requests (work orders), parts replacements and modifications.

The work request form (AP-5, Page 15) contains the classification information, which is derived from the equipment classification list (Q-List) by the work request originator or the approving supervisor. A Quality Assurance representative checks the classification again using the equipment classification list (Q-List) (AP-5, Page 6 and 7).

Maintenance procedures are in the process of being reviewed to assure that any classification information is correct. The review of Maintenance Department maintenance procedures is complete. The review of I&C Department maintenance procedures is ongoing and will be completed prior to startup.

Nine Mile Point Unit 2 has an ongoing program to ensure that vendor information is complete, current and controlled throughout the life of the plant, and appropriately referenced in procedures. This program is conducted in three parts. The first part is the AP-3.4.2, Operations Experience Assessment program which receives, reviews and acts on applicable information from the reactor trip system supplier for Nine Mile Point Unit 2. The information consists of General Electric Service Information Letters (SILs) which the Independent Safety Engineering Group (ISEG) receives and reviews to determine applicability. TDP-5, Administration of Operational Engineering Assessment Items, provides guidance to the ISEG for the handling of OEA items to assure complete and accurate closeout of potential operating problems.

In addition, the Operations Assessment Program addresses information from the Nuclear Regulatory Commission (NRC) such as I&E Notices, Circulars and Bulletins, as well as information from the Institute of Nuclear Power Operations (INPO) such as Significant Event Reports and Significant Operating Experience Reports. Collectively, these sources of information provide a comprehensive and timely mechanism to assure that information pertaining to problems with safety-related equipment are identified and corrected.

Thus, together with our current participation in the General Electric Operations Engineer Program, a high level of communication, feedback and equipment performance improvement is achieved.

2.1 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS)

Position

Licensees and applicants shall confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures and information handling systems used in the plant to control safety-related activities, including maintenance, work orders and parts replacement. In addition, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information is complete, current and controlled throughout the life of the plant, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of these components should be contacted and an interface established. Where vendors cannot be identified, have gone out of business, or will not supply the information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair to compensate for the lack of vendor backup and to assure reactor trip system reliability. The vendor interface program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgement for receipt of technical mailings. The program shall also define the interface and division of responsibilities among the licensees and the nuclear and non-nuclear divisions of their vendors that provide service on reactor trip system components to assure that requisite control of, and applicable instructions for maintenance work are provided.

NMP2 Response

Niagara Mohawk does not currently plan to develop a specific list of components that would comprise a reactor trip system. The reactor trip function is accomplished at Nine Mile Point Unit 2 by utilizing redundant plant process instrumentation that input to a one-out-of-two taken twice logic system. These signals initiate a reactor trip (rapid control rod insertion i.e. scram) by deenergizing solenoid operated scram pilot valves that vent air from the reactor scram valves.

The components that contribute to the reactor trip function are contained in several systems rather than one reactor trip system. Those systems whose components contribute to the reactor trip function include the reactor protection system, reactor vessel instrumentation system, neutron monitoring system and control rod drive system. Therefore, a new system identified as the reactor trip system would cause unnecessary inconsistencies with existing Nine Mile Point Unit 2 system nomenclature. This would require extensive revision to existing documentation and training program with no enhancement of safety.

2.1 (Cont'd)

The second part of this program is the Administrative Control of technical manuals. The current method used to control the flow of technical information is Stone & Webster's Project Procedure PP-81, Method for Handling Supplier Technical Documents. This procedure states a specific program in which technical documents are received and transmitted to the appropriate personnel for proper channeling. All technical documents are received at Stone & Webster's Operations Center where a responsible engineer reviews it to ensure that information is technically adequate and applicable to the equipment purchased. It is then transmitted to the site (Nine Mile Point's Document Control) where it is checked for comments, issued as a controlled document and maintained throughout the life of the plant. This procedure will stay in effect until a similar program, such as the one being implemented at Unit 1, NEL-014G, Control and Distribution of Vendor Documents can be developed.

The third part of the program is Niagara Mohawk's Technical Review and Control of maintenance procedures per Section 6.5.2 of Technical Specifications, which is administered through AP-2, Production and Control of Procedures. This is a unique feature of the Nine Mile Point Technical Specifications which assures that a thorough technical review is performed on all safety-related procedures, rather than a cursory review and approval by the Site Operations Review Committee as could occur at nuclear stations with Standard Technical Specifications.

These three parts provide Unit 2 with an improved method of evaluating and controlling technical information which subsequently enhances Nine Mile's position on safety.

2.2 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (PROGRAMS FOR ALL SAFETY-RELATED COMPONENTS)

Position

Licensees and applicants shall submit, for staff review, a description of their programs for safety-related equipment classification and vendor interface as described below:

1. For equipment classification, licensees and applicants shall describe their program for ensuring that all components of safety-related systems necessary for accomplishing required safety functions are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders and replacement parts. This description shall include:

2.2.1.1 The criteria for identifying components as safety-related within systems currently classified as safety-related. This shall not be interpreted to require changes in safety classification at the systems level.

2.2.1.1 (Cont'd)

NMP2 Response

NMP-2 utilizes the quality group classification system for classifying the water, steam, and radioactive waste containing components important to the safety of water-cooled nuclear power plants. This system established by NRC Regulatory Guide 1.26, "Quality Group Classification and Standards," defines the Quality Group Classification System consisting of four Quality Groups A, B, C, and D. The definition of Quality Group A (Class 1) is provided by 10CFR50.2 (V) under "Reactor Coolant Boundary". The definitions of Groups B, C, and D are provided by Regulatory Guide 1.26.

Niagara Mohawk's architect engineer, Stone & Webster, used this guide to develop a detailed "Equipment and Structure Classification List" located in Section 3.2 (Classification of Structures, Systems, and Components) of the FSAR. This section states that, "Seismic Category I structures, systems and components are necessary to ensure:

1. The integrity of the reactor coolant pressure boundary (RCPB).
2. The capability to shut down the reactor and maintain it in a safe shutdown condition.
3. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR100."

The criteria used for identifying equipment as safety-related on documents, drawings, and information handling systems is Stone & Webster's procedure C-3, Equipment Identification Codes. This procedure describes a format and application by which the equipment is identified in such a manner to allow control during all phases of plant design and construction. Each piece of equipment is identified by an equipment code number. This code number is divided in two by either an asterisk(*) for safety-related equipment, or a dash(-) for all other equipment. This provides a systematic way in which safety-related equipment can be identified by operating personnel in quick concise manner. Therefore, NMP2 meets the intent of Section 2.2.1.1.

2.2.1.2 A description of the information handling system used to identify safety-related components (e.g. computerized equipment list) and the methods used for its development and validation.

NMP2 Response

The current listing of safety-related equipment is provided in FSAR Table 3.2-1. Currently, this document is being used by plant and engineering personnel to identify safety-related components. As mentioned in Section 2.1, a task is currently underway to upgrade the details and accessibility of the equipment classification list. This upgrade is described as follows and will be implemented when the data is fully validated.

The Information Handling System that will be used to identify safety-related components is the Master Equipment List (MEL). The MEL is a computer data base which will ultimately consist of on-line information on all equipment installed at NMP2. This data base forms the nucleus of an information system that ties engineered component attributes to one installed component attributes, two active component documents, three spare parts necessary to maintain components, and four archived component documents. Eventually, it will form an operational authority file which interfaces with other computer data bases which track scheduled and unscheduled maintenance, equipment qualification requirements, in-service inspections, and modifications of plant components, thus ensuring configuration integrity for NMP2 as well as ready access for station supervision.

The MEL was developed from all major existing computerized design information systems on cables, raceways, equipment, pipe lines & supports etc., and then integrated into one data base. The design information provided by the NSSS vendor and A/E was developed from engineering evaluations performed by GE and Stone & Webster engineers using the criteria of FSAR Section 3.2.

Validation of the MEL for safety-related components is accomplished on a system basis by an extensive check of the component identification number against drawings, existing data bases, testing information, name plate serial numbers and if necessary, physical inspection in the plant. This effort is currently continuing.

2.2.1.3 A description of the process by which station personnel use this information handling system to determine that an activity is safety-related and what procedures for maintenance, surveillance, parts replacement and other activities defined in the introduction to 10CFR50, Appendix B, apply to safety-related components.

2.2.1.3 (Cont'd)

NMP2 Response

The following is a description of the process of determining if an activity is safety related. The supervisor of the department responsible for the activity has the responsibility to utilize the Equipment Classification List (Q-List) to determine the equipment classification. Documents such as work requests and purchase requisitions are reviewed and approved by the Quality Assurance Department. Activities such as surveillance or preventative maintenance are covered by procedures which are reviewed and approved per Section 6.5.2 of the Unit 2 Technical Specifications. These attributes are specified in various administrative procedures currently in place. The final administrative control before work occurs is approved by the Shift Supervisor. Based on the training, experience and knowledge of Technical Specifications required to fill the position, the Shift Supervisor can determine if the correct practices are to be used. This control includes sign-offs in the procedures, work requests and markups (tags) to be used. It is the intent of the process at Nine Mile Point Unit 2 to have checks and balances on the system to assure that an error on the part of an individual will not result in "non-safety related practices" being applied to safety-related equipment.

2.2.1.4 A description of the management controls utilized to verify that the procedures for preparation, validation and routine utilization of the information handling system have been followed.

NMP2 Response

Safety-related activities are governed by various administrative controls which implement the Quality Assurance Program. Adherence to the Quality Assurance Program is monitored primarily through the use of audits and inspections. These audits and inspections encompassed the various safety-related activities and are performed at various frequencies. For example, maintenance activities on safety-related equipment are subject to quality assurance inspections on a routine basis. Other audits or inspections are performed less often but cover a longer period of operation or activity. Items of non-compliance identified as a result of these audits and inspections are documented in accordance with provisions of the quality assurance program and are carried as open items until resolved.

2.2.1.5 A demonstration that appropriate design verification and qualification testing is specified for procurement of safety-related components. The specifications shall include qualification testing for expected safety service conditions and provide support for the licensees' receipt of testing documentation to support the limits of life recommended by the supplier.

NMP2 Respnse

The Nuclear Regulatory Commission conducted an audit of the Equipment Qualification program at Nine Mile Point Unit 2. The results of that audit are detailed in letter dated 1-8-85 from Elinor G. Adensam to B. G. Hooten.

2.2.1.6 Licensees and applicants need only to submit for staff review the equipment classification program for safety-related components. Although not required to be submitted for staff review, your equipment classification program should also include the broader class of structures, systems and components important to safety required by GDC-1 (defined in 10CFR Part 50, Appendix A, "General Design Criteria, Introduction").

NMP2 Response

With respect to the equipment classification program in use at Niagara Mohawk for structures, systems and components Important to Safety, we are participating in the Utility Safety Classification Group and are seeking a generic resolution to the Staff's concern in this regard through the efforts of the Group. We do not agree that the plant structure and components important to safety constitute a broader class than the safety-related set. Nevertheless, we believe that non-safety related plant structures, systems and components have been designed and are maintained in a manner commensurate with their importance to the safety and operation of the plant.

<u>Section 3.1 & 3.2</u>

Generic Letter 83-28

Post-Maintenance Testing (Safety Related Systems)

3.1 & 3.2 POST-MAINTENANCE TESTING

Positions

The following actions are applicable to post-maintenance testing:

3.1.1 Licensees and applicants shall submit the results of their review of test and maintenance procedures and Technical Specifications to assure that post-maintenance operability testing of safety related components in the reactor trip system is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

3.2.1 Licensees and applicants shall submit a report documenting the extending of test and maintenance procedures and Technical Specifications review to assure that post-maintenance operability testing of all safety related equipment is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

NMP2 Response

The following is in response to sections 3.1.1 and 3.2.1 and is presented with the intent that NMPC does not distinguish between Reactor Trip System and all other safety-related systems in implementing Post-Maintenance Testing Requirements.

Niagara Mohawk has made improvements to administrative and implementing procedures to more clearly satisfy the Post-Maintenance Testing (PMT) requirements of Generic Letter 83-28. AP-5 "Procedure for Repair" (pages 5,9,10,15 and 18) specifies the requirement for PMT following any maintenance of Safety Related equipment.- TDP-8 "Post-Maintenance Testing Criteria" provides guidance on the type of testing required based on the type of component and the type of maintenance performed.

This process applies to systems that have been turned over to NMPC from Construction and is summarized as follows: The department supervisor receiving the Work Request (AP-5.0, page 15) determines if the departmental procedure for accomplishing the maintenance task, or another departmart('s) procedure, incorporates a maintenance test that meets the requirements given in TDP-8. If so, he denotes the procedure number on the WR (line #15) and on the PMT requirements line (line 37). If not, line #37 is left blank. Upon completion of the work, the WR is returned to the Control Room, where the Station Shift Supervisor or the Assistant Shift Supervisor review the WR including line #37. If the Operations Department has a procedure which meets the testing requirements of TDP-8, it is denoted on line #37, and performed. Successful performance results in the Station Shift Supervisor or Assistant Station Shift Supervisor accepting the system/component for return to service. An unsuccessful test results in the initiation of another WR.

3.1.1 & 3.2.1 (Cont'd)

If no procedure exists for testing the system/component in relation to the maintenance performed, (which could be the case for a safety related component or system that is not in Technical Specifications) a PMT Test Report is completed per AP-5 and attached to the WR. Generally, this will involve placing the component in service and witnessing proper operation.

Further, maintenance procedures which do not contain post-maintenance tests generally contain steps to notify the appropriate department to conduct a test. However, the WR is the administrative control.

Thus, Nine Mile Point Unit 2 is currently in compliance with Post-Maintenance Testing requirements of Generic Letter 83-28.

3.1.2 Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

NMP2 Response

As stated in Section 2.1, the General Electric SIL program constitutes the RTS Vendor Interface Program. The post-maintenance testing recommendations contained in the SILs have been identified and are being handled via the Operations Assessment Program. As of this writing all of these have been assigned to an engineer for incorporation of applicable information. It is expected that disposition of all of these will occur prior to fuel load.

3.1.3 Licensees and applicants shall indentify, if applicable, any post-maintenance test requirements in existing technical specifications which can be demonstrated to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval. (Note that action 4.5 discusses on-line system functional testing.)

NMP2 Response

Technical Specifications have been reviewed for Post-Maintenance Testing Requirements that can be demonstrated to degrade safety rather than enhance it. None were identified.

3.2.2 Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications where required.

3.2.3 (Cont.d)

NMP2 Response

As stated in Section 2.2.2, the General Electric SIL program constitutes the Safety Related Systems Vendor Interface Program. The post-maintenance testing recommendations contained in the SILs have been identified and are being handled via the Operations Assessment Program. As of this writing all of these have been assigned to an engineer for incorporation of applicable information. It is expected that disposition of all of these will occur prior to fuel load.

3.2.3 Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which are perceived to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval.

NMP2 Response

Technical Specifications have been reviewed for Post-Maintenance Testing Requirements that can be demonstrated to degrade safety rather than enhance it. None were identified.

Section 4.5

Generic Letter 83-28

Reactor Trip System Reliability (System Functional Testing)

4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

Position

On-line functional testing of the reactor trip system, including independent testing of the diverse trip features, shall be performed on all plants.

4.5.1 The diverse trip features to be tested include the breaker undervoltage and shunt trip features on Westinghouse, B&W and GE plants; the circuitry used for power interruption with the silicon controlled rectifiers on B&W plants; and the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.

NMP2 Response

Generic Letter 83-28, Section 4.5 recommends on-line functional testing of scram pilot valves and scram backup valves. At Nine Mile Point Unit 2, the scram pilot air system controls and supplies air to operate the scram valves and the scram discharge volume vent and drain valves. The control air is supplied through two backup scram, and two Redundant Reactivity Control System (RRCS) solenoid operated air valves to the scram pilot valves at the individual control rod drive Hydraulic Control Units (HCU) and the scram discharge volume vent and drain valves, per each of two HCU Air Headers. The backup scram valves receive signals from the reactor protection system, as do the pilot solenoids, to each scram, vent and drain valve, providing redundancy and increasing system reliability. In the event that the scram pilot valves fail to function, the action of the backup scram valves assure that the control rods insert, thus, enhancing the reliability of the reactor trip function.

The backup scram valves are normally de-energized, DC solenoid operated valves. When at least one pair of channel sensor relays in <u>both</u> trip systems de-energize (one out of two taken twice logic), both backup scram valve solenoids energize and reposition the backup scram valves to block the instrument air supply and exhaust the scram air header. This action alone will cause the insertion of all control rods. The check valve around backup scram valve B allows the pilot air header to bleed down even if backup scram valve B fails to change position. Thus, the failure of one backup scram valve to operate will not prevent a scram, and the operation of one backup scram valve will cause a scram of the one half of the control rods.

Current testing of the scram pilot valves is accomplished through the existing technical specification surveillance program. The surveillance tests, taken together, functionally test the trip system from the sensing instrument, through the trip logic circuitry, to the scram pilot valves. The surveillance procedures are written to test the one-out-of-two taken twice logic in such a manner that the channels are tested independently. This allows one-half of the necessary logic to "makeup," actuating the entire trip channel up to and including one out of the two scram pilot valves on every control rod's scram inlet and discharge valves.

4.5.1 (Cont'd)

Scram testing is performed during each operating cycle in accordance with Technical Specification. This scram time testing demonstrates the action of the pilot scram valves and scram inlet and discharge valves.

In series with the backup scram valves are two normally deenergized DC RRCS solenoid operated Alternate Rod Insertion (ARI) valves. Similar to the B backup scram valve, each RRCS valve has a check valve in a bypass line so its failure will not prevent the other RRCS or the backup scram valves from depressurizing its scram air header. The ARI function of RRCS is actuated on failure to scram symptoms, i.e. high reactor vessel pressure or low-low reactor water level.

Because of the design of the system, on-line testing of one backup scram or one RRCS valve would result in a full scram of one half the control rods. This would be an unacceptable situation which would result in an automatic or a manual full scram of all the control rods. Therefore, on-line testing of backup scram valves or RRCS valves will not be performed. However, backup scram valves and RRCS valves will be tested on a refueling interval.

A plant specific reliability study was performed by GE in NEDE 22157 for RRCS and ARI. The results of this study showed that these systems are highly reliable.

4.5.2 Plants not currently designed to permit periodic on-line testing shall justify not making modifications to permit such testing. Alternates to on-line testing proposed by licensees will be considered where special circumstances exist and where the objective of high reliability can be met in another way.

NMP2 Response

As described in 4.5.1, Nine Mile Point Unit 2 is not designed for on-line testing of the backup scram or the ARI valves. The current design would result in scram of one half of the rods, if one of the backup scram, or ARI valves were energized while on-line. However, due to the multiple redundancy of the system, ie. the backup scram valves are redundant to the scram pilot valves, and are also redundant to each other, modifications to permit on-line testing are not warranted.

Additionally, the ARI valves are a redundant scram system, utilizing independent sensors from the Reactor Protection System and capable of completing a scram with the total failure of the normal scram system. The ARI valves are controlled by the Redundant Reactivity Control system, which is also redundant.

NMPC endorses the following excerpt from NEDC-30505 "Response Guidelines for NRC Generic Letter 83-28" prepared by General Electric for the BWR Owners Group.

"The Nine Mile Point Unit 2 Reactor Protection System design complies with all applicable regulatory requirements for the RPS.

4.5.2 (Cont'd)

The remainder of this paragraph is a summary of the on-line functional testing and testing intervals performed on the RPS. Consistent with the Technical Specifications, on-line channel functional testing is performed on the multiple and diverse reactor transient trip sensors [Average Power Range Monitor (APRM) and intermediate Range Monitor (IRM) Reactor trip signal channels, and multiple and diverse Scram Discharge Volume High water level trips]. During the required trip sensor channel tests discussed above, each scram contactor which actuates the scram pilot solenoid valves is tested. The simple operation of the scram contactors minimizes concerns of wear, and frequent testing assures that any failures are detected early. The Scram Pilot Solenoid Valves which are actuated by the scram contactors are all tested regularly. Redundant Electrical Protection Assemblies (EPAs) which protect the Scram Pilot Solenoid Valves from low voltage chattering (and the associated potential consequence of accelerated wear) are also functionally tested. These surveillance testing requirements related to the Scram Pilot Solenoid Valves assure that the probability of undetected failure of these solenoid valves is small. In summary, the current RPS on-line surveillance requirement, in conjunction with multiple and diverse scram sensors, assure that the probability of failure of enough control rods to prevent scram is negligible.

Channel functional tests are performed on-line for the following sensor trips:

- Reactor Vessel Dome Pressure-High
- Reactor Vessel Water Level-Low
- Main Steam Line Isolation Valve-Closure
- Main Steam Line Radiation-High
- Drywell Pressure-High
- Turbine Control Valve Fast Closure, Control Oil Pressure-Low
- Turbine Stop Valve-Closure

Channel functional tests are also performed for APRMs and IRMs.

In References 1 and 2, it is shown that each of the above plant variables used to initiate a protective function is backed up by a completely different plant variable. In fact, it can be seen from Table 1 that for the most frequent transients, scram is initiated by three diverse sensors in all but one case (regulator failure-primary pressure increase which is intitiated by two diverse sensors). This indicates that adequate redundancy exists in the design to provide protection against multiple independent sensor failures. Also, diversity among sensor types reduces the potential for common cause failures, failures due to human error, and increases in failure rate due to wearout. A pictorial representation of the RPS logic configuration is provided in Figure 1.

4.5.2 (Cont'd)

Each sensor channel functional test includes full actuation of the associated logic, the two output scram contactors in each channel, and the individual CRD scram air pilot valve solenoids for the associated logic division (solenoids from both logic Division A and B are required for scram initiation).

The most credible failures within the RPS logic will de-energize a set of scram solenoids which causes a half scram, i.e., one of the two scram solenoids required for scram initiation is de-energized at some or all hydraulic control units. These failures would be "SAFE" failures that would increase the probability of plant shutdown.

The less credible logic failures which prevent a channel from de-energizing will be detected during channel functional test in compliance with Technical Specification requirements. The tests described above ensure that an increase in failure rate due to a wearout condition or a common cause failure potential could be detected early and corrective action taken before the failure condition becomes systematic.

Other channel functional tests include testing of the Scram Discharge Volume (SDV) Water Level-High trip and manual scram trip and test of the reactor mode switch in the shutdown position every refueling. The first two trips involve on-line testing and the latter mode switch test can only be conducted during reactor shutdown. The manual scram trip can be tested on-line without creating a scram.

The testing of the SDV Water Level-High trip is considered adequate based on the current designed redundancy and diversity incorporated into the system. There are two diverse and redundant sets of level sensors which scram the reactor in the unlikely event of high water level in the SDV during power operation. These trips are designed to allow sufficient scram water discharge volume given the scram trip point is reached.

Reference 2 concluded that reactor shutdown can be achieved if at least 50% of the control rods in a checkerboard pattern and 69% in a random pattern are inserted in the core. The probability of independent failure of enough rods to prevent shutdown is negligible. The most unlikely type of failure would be some common cause mechanism that if undetected over a long period of time would cause unsafe shutdown. The Technical Specification surveillance requirements adequately ensure that a failure mechanism affecting several individual drives (considered to be very remote) would not go undetected. One of the major features that ensures that several drives do not fail at one time due to wearout or a common cause is the staggered maintenance and overhaul of selected degraded CRDs or Hydraulic Control Units (HCUs) at refueling outages. This ensures a mix of drives by age, component lot, maintenance time and servicing personnel, and testing.

4.5.2 (Cont'd)

The scram insertion time tests include, in addition to drive timing and insertion capability, a test of operability of the HCU scram insert and discharge valves including associated scram air pilot valves. As stated in the previous paragraph, the required frequency of testing given in the Technical Specification ensures that a systematic failure mechanism in the HCUs would be detected early enough and corrective action taken before the condition becomes a critical failure preventing scram."

Therefore, since the scram pilot valves are tested weekly during APRM half scram tests, and since the backup scram valves and the ARI valves will be tested once a refueling cycle, and since rod scram time testing is performed at on a refueling cycle or more frequently in accordance with Standard Technical Specifications, on-line testing of the backup scram and ARI valves is not warranted.

4.5.3 Existing intervals for on-line functional testing required by Technical Specifications shall be reviewed to determine that the intervals are consistent with achieving high reactor trip system availability when accounting for considerations such as:

1. uncertainties in component failure rates
2. uncertainty in common mode failure rates
3. reduced redundancy during testing
4. operator errors during testing
5. component "wear-out" caused by the testing

Licensees currently not performing periodic on-line testing shall determine appropriate test intervals as described above. Changes to existing required intervals for on-line testing as well as the intervals to be determined by licensees currently not performing on-line testing shall be justified by information on the sensitivity of reactor trip system availability to parameters such as the test intervals, component failure rates, and common mode failure rates.

NMP2 Response

Nine Mile Point Unit 2 on-line functional testing and testing intervals are performed consistent with the Technical Specifications which are based on Standard Technical Specifications. The following reactor trips are functionally tested on-line.

Manual Scram
High Reactor Pressure
High Drywell Pressure
Low Reactor Water Level
High Water Level Scram Discharge Volume
Main Steam Line Valve Position
High Radiation Main Steam Line
Neutron Flux
  Intermediate Range Monitor (IRM) (when required)
  Average Power Range Monitors
Turbine Valve Closure
Generator Load Rejection

4.5.3 (Cont'd)

In addition, the shutdown position of the reactor mode switch scram function is tested during refueling outages. During the testing discussed above, the scram pilot solenoid valves are tested, in that one of the two scram pilot valves on every control rod scram inlet and outlet valves are activated. Also, overvoltage, undervoltage and underfrequency protection is provided for the reactor trip bus including power to the scram pilot valves.

For the major transients evaluated, the number of independent scram features which are available to terminate a particular transient are listed in the response to Section 4.5.2 above. Therefore, it can be demonstrated that adequate redundancy exists in the Nine Mile Point Unit 2 design to provide protection against multiple independent sensor failures.

Further, NMPC participated in and endorses the "BWR Owners Group response to NRC Generic Letter 83-28, Item 4.5.3" NEDC-30844. This document contains analyses performed by General Electric that concluded that the current on-line functional testing intervals are adequate to achieve high reactor trip system availability.

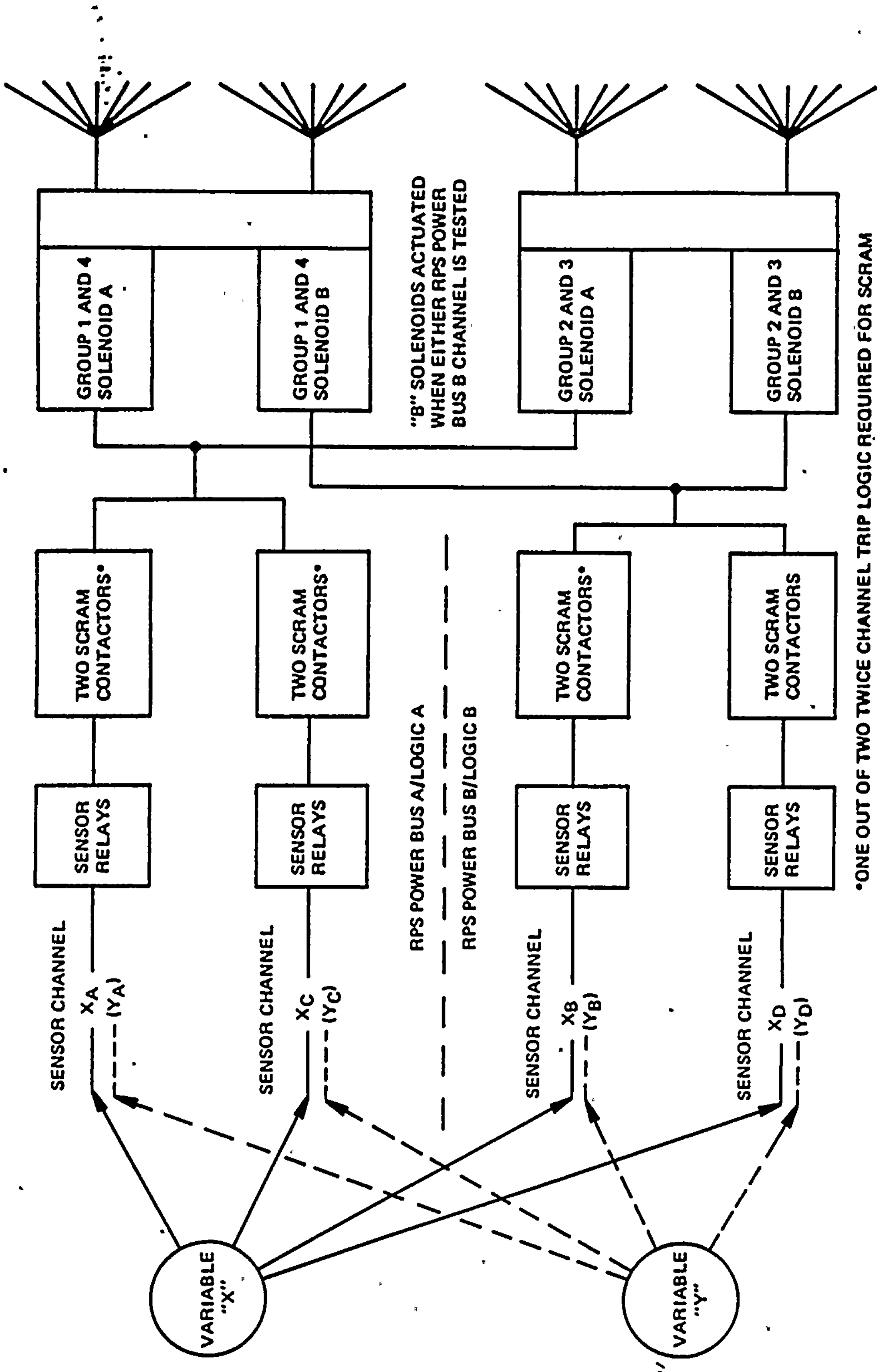
In summary, the current reactor protection system on-line surveillance program requirements, in terms of scope and testing intervals, in conjunction with multiple and diverse scram sensors assures the probability and reliability of the reactor trip system to function to effect control rod insertion and resulting reactor shutdown.

Further, for Unit 2 an automatic standby liquid control system is installed which provides redundant means to shut down the reactor.

Table 1
SENSOR DIVERSITY FOR MAJOR TRANSIENTS

Transient	Scram Signals - Order of Occurrence						
	Inputs From Pressure or Differential Pressure Transmitters and Trip Units		Inputs From Pressure Position or Micro Switch Contact Opening			Inputs From Neutron Flux or Radiation Sensors	
	Reactor Pressure >1065 PSIG	Reactor Level <Level 3	Turb Cont. Valve Oil Pres. Set Pt.	Turb Stop Valve Pos <90% Full Open	MSIV Pos. <90% Full Open	APRM >120%	MSIV Hi Rad. >6 x Background
MSIV Closure	3	4			1	2	
Turb Trip (with bypass)	3			1		2	
Generator Trip (with bypass)	3		1			2	
Pres. Regulator Failure (primary pressure decrease)	3	4			1	2	
Pres. Regulator Failure (primary pressure increase)	2					1	
F.W. Flow Control, Failure (reactor water inventory increase)	3			1		2	
F.W. Flow Control, Failure (reactor water inventory decrease)	3	1			2		4
Loss of Condenser Vacuum	3		4	1	5	2	
Loss of Normal AC Power	4	5	2	1	6	3	

Figure 1 RPS Relay Logic Configuration



REFERENCES

1. NEDO-1-189, "An Analysis of Functional Common-Mode Failures in GE BWR Protection and Control Instrumentation," L. G. Frederick, et al, July 1970.

2. "BWR Scram System Reliability Analysis," W. P. Sullivan, et al, September 30, 1976 (Transmitted in letter from E. A. Hughes (GE) to D. F. Ross (NRC), "General Electric Company ATWS Reliability Report," September 30, 1976).

Enclosure 3

(1)

Nine Mile Point, Unit 2, Response Survey 3/31/86.

1. Item 2.1: (a) Part 1: The response is partially complete. Those components that function to provide RT should be reviewed to determine that all such components are classified as safety-related on all relevant plant documents. (b) Part 2: The program and its goals for vendor manuals should be briefly described. Also the permanent update program should be described.

2. Item 2.2.1: Response appears complete.

2.2.1.1: Response appears complete, but should address subcomponents and parts.

2.2.1.2: Response should state how the Q List was prepared; how new items are entered; how changes in classification of listed items are made; how listed items are verified; how unauthorized changes to the listing are prevented; and how the listing will be maintained and made available to users as an official, single, consistent, and unambiguous version.

2.2.1.3: Response appears complete.

2.2.1.4: Response is not complete:
(a) Administrative controls should be identified.
(b) Application of the QA program to preparation, maintenance, and usage of the Q List to identify safety-related equipment in maintenance, test, operating, and parts procurement procedures should be described.

(2)

2.2.1.5 : Response states that Engineering Procedures will address these requirements, however (a) Have these procedures been written? (b) If not, provide schedule for their completion, and (c) They should require that replacement parts meet the same specifications as the original.

2.2.1.6 : Response appears complete.

3. Item 2.2.2 : Response is incomplete. The response should address these items :

4/10/84
[illegible]

(a) Guideline (Accomplishment of Program Enhancements): based on AP 3.4.2 and TDP-5

~~Licensee~~ Applicant's response should briefly describe how their program will accomplish the program enhancements recommended in Section 3.2 of the NUTAC/VETIP Report (INPO 84-011). These include:

- For NPRDS:

(1) Improvement of the definition of component to better describe components other than electrical.

(2) Improve the failure reporting guidance to:

- provide better information on the role played by piece parts in causing component failures;

- recommend that utility provide information when inadequate vendor information is a causal or contributing factor in a failure.

- recommend that utility provide more complete information including details of failure analyses in the failure reports.

- assure that utilities submit preliminary reports promptly, and for complicated failure cases, upgrade the initial report with later, more detailed supplementary reports. NPRDS procedures should be revised to recommend this procedure.

- increase the scope of NPRDS reporting to allow more comprehensive and complete reporting.

- provide guidance that requires utilities to report when inadequate, faulty, or missing maintenance or test procedures are contributing factors in a failure.

For SEE-IN:

Reports should be generated for potential failures caused by faulty or missing vendor supplied information or other ETI. Such occurrences should be reported over NUCLEAR NETWORK.

4

b) Guideline (NUTAC/VETIP Implementation Responsibilities):

Licensee response should describe briefly how their program will accomplish the implementation responsibilities recommended in Section 4.1.1 of the NUTAC/ VETIP Report. These include:

- Establishment and maintenance of vendor interface with NSSS supplier.
- Have a program of seeking assistance from other vendors of safety-related equipment when found necessary.
- Have procedures for processing all incoming ETI regardless of source to assure prompt review, evaluation, and distribution of results so that
  - (1) Key personnel are promptly warned of possible problems;
  - (2) New or revised information is incorporated into plant procedures and programs;
  - (3) Significant ETI is shared with other utilities via NUCLEAR NETWORK reports;
  - (4) administrative procedures should require that plant procedures at least reference appropriate ETI;
  - (5) Appropriate ETI should be incorporated into the performance and quality review sections of safety-related procedures;
  - (6) Vendors or outside contractors who perform or provide safety-related services shall be subject to adequate utility control and shall conform to utility or utility-approved QA procedures and controls.

5

c) Guideline (Vendor Contact):

Licensee response should show that interfaces have been or are being established with at least two or more major vendors of safety-related equipment other than the NSSS. Examples of such vendors include: diesel generator vendor, switchgear vendor, major pumps vendor, or vendor of motor-operated valves.

d) Guideline (Commitment To Aid In Accomplishment Of Implementation Responsibilities):

Licensee response should show that they have committed to work with INPO to ensure accomplishment of INPO Implementation Responsibilities as described in Sections 3.2, 4.1.2, and 4.2.2.1 of the NUTAC/VETIP report.

e) Guideline (NSSS Vendor Contact):

The vendor interface program should include periodic contact with the NSSS vendor to assure that the latest versions of maintenance, test, service, and modification recommendations are in the licensee's possession.

f) Guideline (Vendor Contact):

The licensee should show that contact has been attempted with major vendors of their safety-related equipment other than the NSSS to establish continuing, periodic interfaces with them for exchange of service, test, maintenance, and modification information. Evidence of such attempts and their results should be retained for audit.

g) Guideline (Assurance of Receipt of ETI):

The vendor interface program should use a system of positive feedback such as licensee acknowledgement of receipt of technical information mailings to assure that licensee has received all current information.

h) Guideline (Division of Responsibility):

Program description shall define the interface and describe the division of responsibilities among the licensee and the nuclear and non-nuclear divisions of their vendors that provide service on safety-related equipment. This is interpreted to mean that the licensee shall remain responsible for controlling the content and application of procedures, instructions, and quality assurance activities to maintenance, test, service, and modification work on safety-related equipment performed by other than licensee organizations and personnel.

4. Item 3.1.3: Commitment appears OK. Applicant should inform when T.S. review is complete for this item

5. Item 3.2.3: As for Item 3.1.3.

6. Item 4.5.2: Response is incomplete. Response states that it will be provided prior to startup. A schedule for submission of response promptly should be provided. The response of 12/20/85 states that backup scram valves will be tested at refueling outages appears acceptable.

7. Item 4.5.3 - Response is incomplete. It states that applicant is participating in a BWROG committee to address Technical Specification Improvements and that a response will be provided prior to startup. It is not clear that this BWROG committee will address the issue and a schedule that provides for prompt submittal should be provided followed by the submittal.

Enclosure 4

Attendance

Name	Organization
Mary Haughey	NRC
David Shum	FOB/NRC
Joel Kramer	EICSB/PWRA
Tony Zallnick	NMPC - Licensing
Robert Randall	NMPC - Generation
T. Loomis	NMPC - Nuclear Licensing
D. R. Lasher	NRC
A. L. Toalston	NRC

April 17, 1986

MEETING SUMMARY DISTRIBUTION

Docket No(s): 50-410
NRC PDR
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ACRS (10)
Project Manager M. Haughey
E. Hylton

NRC PARTICIPANTS

M. Haughey
D. Shum
D. Lasher
A. Toalston

bcc: Applicant & Service List