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VASSALLO,D.B. Operating Reactors Branch 2

SUBJECT: Forwards response to NRC 831019 request for addl info re generic implications of ATWS events (Generic Ltr 83-28), describing scheduling & status. Info should be considered preliminary. Util participating in industry-wide groups.

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RECIP.NAME RECIPIENT AFFILIATION
VASSALLO,D.B. Operating Reactors Branch 2

SUBJECT: Forwards response to NRC 851019 request for addl info re generic implications of ATWS events (Generic Ltr 83-28), describing schedule & status.Info should be considered preliminary.Util participating in industry-wide groups.

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	NRR/DL/ORAB	1	0	NRR/DL/TSRG	1	1
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NIAGARA MOHAWK

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

November 8, 1983

Director of Nuclear Reactor Regulation
Attention: Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

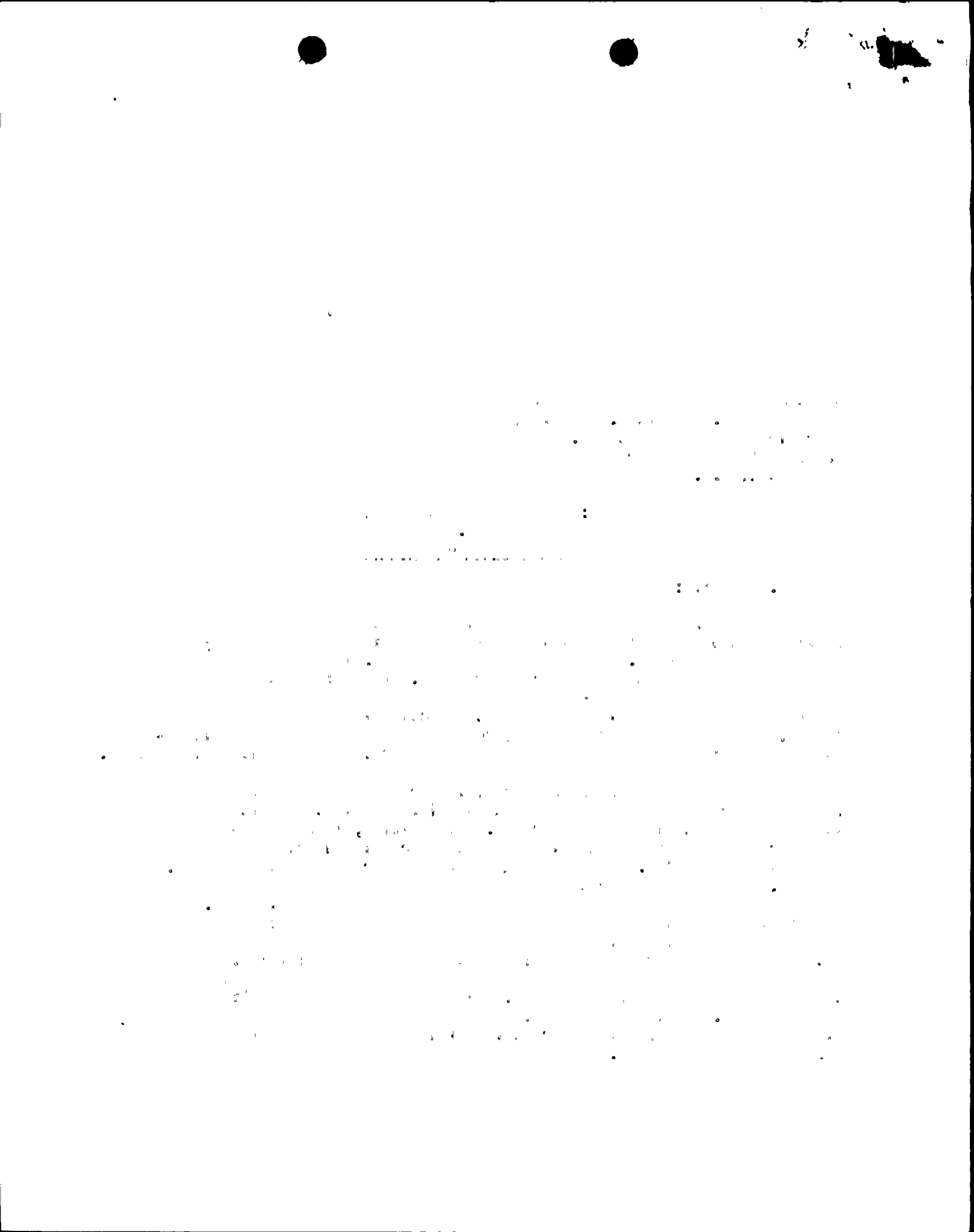
Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Dear Mr. Vassallo:

Your July 8, 1983 letter requested information regarding the generic implications of the Salem Anticipated Transient Without Scram (ATWS) Events (Generic Letter 83-28). Our letter of September 6, 1983 requested a delay in providing the information until February 1984. The delay was requested to allow formation of industry-wide groups and where appropriate, to formulate generic programs addressing the regulatory action items of the generic letter. These industry groups are currently developing programs which may be incorporated in the existing administrative controls at Nine Mile Point Unit 1.

Although our letter of September 6, 1983 requested a revision to the information request schedule, your October 19, 1983 letter indicated a schedule change could not be approved. In addition, it requested information on our current status and any improvements currently anticipated with appropriate schedules. Attachment A provides the requested information. However, because the generic efforts of industry groups are currently in progress, the information provided should be considered preliminary. In addition, the applicability of these activities to the existing Nine Mile Point Unit 1 administrative controls will have to be evaluated prior to any decisions to incorporate the results of these generic activities. It is currently anticipated the efforts of these industry groups will either be completed or underway with a specific completion schedule established by February 1984. Following appropriate review of the results of these efforts, Niagara Mohawk will finalize our action plan and provide additional information at that time.

A001
1/1



Due to the relatively short time provided by your October 19, 1983 request, Niagara Mohawk has not been able to complete its internal verification process with regard to the contents of the letter. This verification will continue and any changes required will be provided.

Sincerely,

T. E. Lempges
Vice President
Nuclear Generation

TEL/RJP:bd
Attachments

ATTACHMENT A

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.

Response

The administrative controls currently being implemented at Nine Mile Point Unit 1 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.

The post trip review at Nine Mile Point Unit 1 is conducted by use of Procedure N1-RAP-6 Post Reactor Scram Analysis and Evaluation (Attachment 1). This procedure requires review and evaluation of specific parameters associated with a reactor scram, and collection and preservation of the data.

Operating Procedure N1-OP-43 Startup and Shutdown Procedure requires the use of the scram report (Attachment 2), pre-startup checkoff sheets (Attachments 3, 4, and 5) and authorization from the Station Superintendent for restart.

Reactor Analysis Procedure N1-RAP-6 and Operating Procedure N1-OP-43 provide the means to evaluate plant transient response and check safety systems for their performance during the transient.

The Reactor Analysis Procedure N1-RAP-6 analysis is conducted by members of the Reactor Analyst Department, and the evaluation is reviewed and approved by either the Reactor Analyst Supervisor or his alternate the Unit Reactor Analyst Supervisor and the Operations Supervisor. The scram report is completed by the Station Shift Supervisor and reviewed and approved by the Operations Supervisor and the Station Superintendent.

1.1.1 The criteria for determining the acceptability of restart.

Response

Although not currently stated in any procedure, the Station Superintendent uses the following criteria for evaluating the acceptability of plant restart following an unscheduled reactor shutdown:

7.1.1 Response (cont'd)

1) The root cause of the reactor scram is known.

2) The plant transient response, including safety system operation has been evaluated and is understood.

3) Any corrective actions that are required prior to a safe restart are completed.

4) Any corrective actions which might reduce the probability of recurrence are identified.

While the above is not specifically stated in the existing administrative controls, it is implicit in the Nuclear Generation Department mission and responsibilities as given below. These criteria are contained in the Niagara Mohawk Corporate Strategic Functional Plan as follows:

Mission:

Safely and efficiently operate and maintain the nuclear power station.

Responsibilities:

The primary responsibilities of the Nuclear Generation Department are as follows:

1. Test and accept the new nuclear power unit and all station modifications and additions.

2. Obtain and train employees to assure safe and efficient startup, operation, and maintenance of nuclear power units.

3. Develop, maintain and monitor procedures to assure safe, reliable and efficient station operation and maintenance.

4. Monitor equipment condition and performance to assure reliability and optimum efficiency. Recommend and implement preventative actions and improvements as required.

5. Assure compliance with license and regulatory requirements.

6. Implement corrective maintenance and modification programs.

7. Provide senior management with requirements for budget, physical plant and personnel needed to fulfill the department mission and responsibilities.

In addition, Niagara Mohawk is currently participating in a Boiling Water Reactor Owners Group effort which will provide generic criteria for plant restart. This criteria will define how to assess safety-related system/equipment operability following an unscheduled shutdown.

1.1.1 Response (cont'd)

Therefore, while currently meeting the intent of Section 1.1.1, appropriate procedures will be reviewed and any revisions deemed necessary will be incorporated. It is anticipated that this effort will be completed approximately one month after the completion of the Boiling Water Reactor Owners Group effort which is currently scheduled for February 1984.

1.1.2 The responsibilties and authorities of personnel who will perform the review and analysis of these events

Response

The general responsibilities and authorities of the Station Superintendent, Operations Supervisor and Station Shift Supervisor are stated in Administrative Procedure APN-2A Conduct of Operations and Composition and Responsibilities of Station or Unit Organization. The responsibilities and authorities of the Reactor Analyst Supervisor and Unit Reactor Analyst Supervisor are stated in Administrative Procedure APN-2A and APN-2 Composition and Responsibilities of Site Organization.

Our initial review indicates that Nine Mile Point Unit 1's Administrative Controls currently meet the intent of Section 1.1.2. However, appropriate procedures will be re-reviewed and any revisions deemed necessary will be completed in conjunction with the efforts associated with item 1.1.1.

Specific responsibilities relative to post-term review are discussed in the response to item 1.1.1 above.

1.1.3 The necessary qualifications and training for the responsible personnel

Response

The qualifications and training for the positions of Station Superintendent, Operations Supervisor, Reactor Analyst Supervisor and Unit Reactor Analyst Supervisor comply with the requirements of ANSI/ANS 3.1-1981. Additionally, the current site Reactor Analyst Supervisor and the current Unit Reactor Analyst Supervisor both hold senior reactor operator licenses.

The qualification and training for the Station Shift Supervisor meets ANSI/ANS 3.1-1981 with the utilization of the Shift Technical Advisor. In addition, we are participating in a Boiling Water Reactor Owners Group effort which will identify the personnel qualification and training requirements used in the nuclear industry. The adequacy of these requirments will be evaluated.

1.1.3 Response (cont'd)

Our initial review indicates that the existing Nine Mile Point Unit 1 administrative controls currently meet the intent of Section 1.1.3. However, appropriate procedures will be re-reviewed and any revisions deemed necessary will be incorporated. It is anticipated that this effort will be completed one month after the completion of the Boiling Water Reactor Owners Group effort, currently scheduled for February 1984.

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).

Response

The sources of plant information necessary to conduct the review and analysis are given in Reactor Analyst Procedure N1-RAP-6 and Operating Procedure N1-OP-43 (Attachments 1 through 5) and in the discussion and attachments addressed in item 1.2.

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)

Response

As stated in Section 1.1.3, the individuals responsible for the event analysis are qualified per ANSI/ANS 3.1-1981 and currently hold senior reactor operator licenses. At their disposal are records of previous reactor trips and Final Safety Analysis Report data which are used at their discretion for comparing the transient to expected reponses.

Therefore, Nine Mile Point Unit 1 currently meets the intent of this section. Procedures will be revised to specifically state the methods and criteria by February 1984.

1.1.6 Criteria for Determining the Need for Independent Assessment.

Response

The criteria is not specifically stated, but the intent of the Nine Mile Point Unit 1 administrative procedures are followed. If the evaluation of Reactor Analyst Procedure N1-RAP-6 and/or the scram report and/or the startup checkoff sheets reveal a condition that is not understood, the Station Superintendent will not authorize restart. Instead, appropriate staff members will be called in to assist in the evaluation of the event.

Normally the Site Operations Review Committee reviews the scram reports following plant restart. In the event that the scram was not fully understood and the staff exhausted available sources of information, Site Operations Review Committee would review the scram report prior to the authorization for restart.

Therefore, Nine Mile Point Unit 1 currently meets the intent of this section. Procedures will be revised to specifically state the criteria by February 1984.

1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY

Position

Licensees and applicants shall have or have planned a capability to record, recall and display data and information to permit diagnosing the causes of unscheduled reactor shutdowns prior to restart and for ascertaining the proper functioning of safety-related equipment.

Adequate data and information shall be provided to correctly diagnose the cause of unscheduled reactor shutdowns and the proper functioning of safety-related equipment during these events using systematic safety assessment procedures (Action 1.1). The data and information shall be displayed in a form that permits ease of assimilation and analysis by persons trained in the use of systematic safety assessment procedures.

A report shall be prepared to describe and justify the adequacy of equipment for diagnosing an unscheduled reactor shutdown.

Response

Adequate data handling information systems are installed and operational at Nine Mile Point Unit 1. The capability to record data and display this information exists. The technical expertise to analyze the sequence of event information (event triggered), post-trip logging data and time history trend information for analog and digital plant parameters also exists.

1.2.1 Capability of Assessing Sequence of Events (on-off indications)

1.2.1.1 Brief description of equipment (e.g., plant computer, dedicated computer, strip chart)

Response

A computer replacement at Nine Mile Point Unit 1 from GEPAC 4060 equipment to a Honeywell 4400 Process Management System has recently been completed. The new system consists of high accuracy, high speed floating point hardware, 128K word Core Memory (800 nanoseconds), high speed large core storage bulk memory, disc memory and color controlled Cathode Ray Tubes. Keyboards are available in the Control Room, Technical Support Center and Emergency Operations Facility along with hard copy fast speed printers.

Additionally, the Nine Mile Point Unit 1 control room contains several dual speed strip chart recorders (Attachment 6) and data loggers to provide redundant plant parameter information for time history and event triggered information. The strip chart recorders are General Electric dual pen recorders, Leeds and Northrup strip recorders, or Tracor Westronic strip chart recorders. The data loggers in use are Accurex Autodata 10 and Accurex Autodata 5 used for specific plant monitoring functions. Logs are generated from all of the above described equipment.

The plant computer functions on a real time basis, handling several activities simultaneously. When an event is triggered (based on predefined parameters) three automatic reports are generated. A sequence of event log, a post-trip log and an alarm history are all printed.

Additionally, this digital data can be correlated to the analog information available with two time history recorder schemes to be discussed in Section 1.2.2. Specialized accoustic monitoring equipment provides immediate response to safety and relief valve actuation which is then logged by the plant computer.

1.2.1.2 Parameters monitored

Response

Sequence of event data (Attachment 7) is identified by computer points W000 through W107. These points reflect scram signals from the Reactor Protection Systems and Turbine and Generator trip parameters. Also contained are isolation signals from major isolation valves, electrical breaker status and safety and relief valve positions.

Post-trip log data (Attachment 8) consists of two logs, a nuclear steam supply system (NSSS) log and a balance of plant (BOP) log. Attachment 8 lists the parameters currently output from these logs and the format.

1.2.1.2 Response (cont'd)

Alarm (contact closure) data on digital points is also generated on the printer for an alarm history capability of an additional 2000 computer points reflecting plant parameters. These are monitored via the normal digital fast scan program with computer points A000-Z199.

1.2.1.3 Time discrimination between events

Response

Sequence of events parameters are based on hours, minutes and seconds cycle. Resolution is approximately 2 milliseconds. Normal digital scan points are based on hours, minutes and seconds with resolution to within one second. Post-trip data is based on variable scan rates from once per second to once every 30 seconds (Balance of plant (BOP) high end = 30 second, Nuclear Steam Supply (NSS) high end = 5 seconds). Single point and group scan of digital groups identifies major blocks of points in alarm.

1.2.1.4 Format for displaying data and information

Response

Sequence of Events format on the Utility #1 printer:

TIME CYCLE PT ID PT DESCRIPTION ALARM STATUS

Digital Fast Scan format on the alarm printer is:

TIME ALARM SYMBOL PT ID PT DESCRIPTION ALARM STATUS

Post-trip log format is displayed on the Utility #2 printer in the Control Room:

Twelve Point Identifications (PT IDs) and their values in 3 columns for each time interval -

TIME 1.....12 PT ID
ROW 1 -3
1.....12 Values

Format for strip recorders is: x axis - plant parameter value
y axis - TIME value with intervals defining time increment

1.2.1.5 Capability for retention of data and information

Response

Data typically cannot be printed as fast as the real time event. However, the data remains in storage buffers until processed for printing. The latest contact status is processed either immediately by priority interrupts and immediately printed (sequence points) or retained until the next scan (in one second) by digital fast scan. Whereas digital data is retained in buffers, the history data related to contact status is time based and retained on bulk storage devices (LCS, Disc, etc.). Hardcopy information is retained in Station Records for up to seven years (strip charts, logs, printouts).

1.2.1.6 Power Source(s) (e.g., Class IE, non-Class IE, non-interruptable)

Response

The Honeywell 4400 computer system is fed from Motor Generator Set 167. This receives its normal power from either powerboard 17B (600 VAC) or battery board 12 (125 VDC). Powerboard 17B is fed from Powerboard 103 (4160 VAC) which receives its power from either offsite (115 kVAC) or Diesel Generator 103. Motor Generator Set 167 could also receive power from the other power train of Powerboard 16B (600 VAC), battery board 11 (125 VDC), Powerboard 102 (4160 VAC) and Diesel Generator 102. Therefore, power reliability is very high.

The Accurex data loggers receive power from Motor Generator Sets 162/172 redundantly and these follow similar power train configurations of the plant process computer.

Strip chart recorders are fed from Instrument and Control Bus 130 which is fed by powerboard 13 (600 VAC) but can be switched manually to power panel 167A.

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

Response

Sufficient data handling information systems are installed and operational at Nine Mile Point Unit 1. Assessment of analog variables is available.

1.2.2.1 Brief description of equipment (e.g., plant computer, dedicated computer, strip charts)

Response

Section 1.2.1.1 identified a Honeywell Project Management System computer system, an Accurex data logger, strip chart recorder and customized hardware for safety/relief valve monitoring.

The analog scanner subsystem of the Honeywell 4400 computer handles approximately 1000 analog inputs. These inputs are then converted to a digitally sampled signal with engineering units for the several types of inputs (flows, thermocouples, RTD, linear analogs and nonlinear analogs (logarithmic). Averaging, rate of change, digital filtering and conversion to calculated values based on multiple analog inputs are also available. High and low alarm limits, variable alarm limits, open thermocouple detection, contact cutout based on analog signals, reasonable high/low signal limits and signal deadband features are normal functions used on each analog input.

Several printed logs and video pages are used for post-trip analysis purposes. These include balance of plant and Nuclear Steam Supply System post-trip logs, alarm history displays and logs, control rod position displays and NSSS logs, and customized trend history displays for selected analog points. In addition, balance of plant performance calculations based on analog inputs and General Electric - Nuclear Steam Supply System thermal/hydraulic and heat balance calculations run automatically and on demand to conclude and summarize post event (reactor/turbine) conditions.

Post-trip logs, control rod/Local Power Range Monitors (LPRM) logs, and plant performance logs output automatically to preselected printers. Alarm history display pages with the ability to page backward or forward through analog values after an event, makes it convenient to assess post-trip conditions. Special logs (22 logs with 20 points per log) are available to assist the operator in assessing post-trip conditions. Specially designed custom software addressing analog value historical recording (up to 100 points) allows one to retrieve time plots of preselected values.

Time history information on plant parameters is easily interpreted by the many strip chart recorders identifying the engineering units range and time scale for the entire transient or event. Data links exist interfacing the accoustic monitoring vibration signals, the accurex fuel zone water level and torus temperature parameters to the plant process computer. From the process computer time history information would be available for the multi-functioning data loggers.

1.2.2.2 Parameters monitored, sampling rate, and basis for selecting parameters and sampling rate

Response

a) Post-trip logs - Balance of Plant. This monitors up to 100 points - five minutes before an event and 30 minutes after an event. Although these analog points are scanned at variable rates of 1, 5, 15, 30 and 60 seconds, the output of these points on this log is at an interval of 30 seconds. Attachment 8 identifies the points on this log. Analog parameters on this log are specifically selected to give an overall picture of plant conditions throughout the heat balance cycle of the reactor and turbine-generator output subsystem.

b) Post-trip log - Nuclear Steam Supply System (NSSS). This monitors up to 10 points five minutes before an event and 30 minutes after an event. Resolution is every five seconds and is outputted automatically upon initiation from the scram signals within the Reactor Protection System and Turbine Generator trip parameters.

The alarm history log and alarm display video provide a real time sequentially triggered output of each analog actual value as well as its high and low alarm limits for easy interpretation of trip values before and after an event. The video paging system presently handles 10 pages of 40 points per page. The alarm history buffer that stores alarm data prior to outputting to a printer has a limited storage space. However, the speed of the computer system and the printers made this an unlimited storage space based on previous plant events (i.e. scram, etc.) experiences.

Attachment 6 identifies the plant parameter and its engineering unit range along with its hardware type of strip chart recorder. Major plant parameters are recorded in hard copy and are retained in the plant archive for historical purposes.

The analog trend history subsystem allows up to 100 points to be monitored and stored for a 30-day period at intervals from once per second to once per hour. Time plots of the parameter for the past 30 minutes is available if the parameter has been preselected for this speed of historical recording. Presently all recirculation pumps and their associated parameters are a part of this system. This trend history application package is 70 percent complete but it requires the development of software to allow easy operator initiation and recall.

The Accurex data logger provides full range fuel zone water level monitoring and torus temperature data via a data link to the plant computer. It also provides local display and hardcopy printout.

Calculated values or composed values are developed from single or multiple analog signals which are converted to appropriate values on a ten minute cycle within the performance calculation software package. These values provide significant information of operational and post operational plant systems.

1.2.2.2 Response (cont'd)

Nuclear Steam Supply System programs operate periodically during plant operation, but are significant to on demand requests to assess rod position and thermal/hydraulic conditions after an event.

1.2.2.3 Duration of time history (minutes before trip and minutes after trip)

Response

Post trip - Balance of Plant	-	5 minutes before, 30 minutes after
Post trip - Nuclear Steam Supply System	-	5 minutes before, 30 minutes after
Alarm history	-	real time based on analog scan rate (0-60 seconds)
Alarm display	-	real time based on analog scan rate (0-60 seconds)
Analog time trends	-	based upon demand for previous 30 minutes of data
Nuclear Steam Supply System	-	on demand with instantaneous calculations
Performance logs	-	calculations every 10 minute update or on demand
Strip chart readings	-	real time based on analysis of chart paper
Rod position logs	-	based on demand requests anytime

1.2.2.4 Format for displaying data including scale (readability) of time histories

Response

a. Post trip logs (Balance of plant/Nuclear Steam Supply System) - See Attachment 8 for formats

b. Alarm history formats -

Time, Symbol, Point Identification Point Description, Actual Value, High limit, Low limit

c. Time trend history -

x, y plot with x as time plot and y as the value.

d. Strip chart recorder data as x, y plot with x as the engineering unit scale and y as the time scale with predetermined intervals. Each recorder can be placed in fast or slow speed.

e. Data logger hardcopy is variable formatted based on predesignated register data - Channel 1 yields Temperature No. 1, etc.

'1.2.2.4 Response (cont'd)

f. Special logs (up to 22 logs) -

Row 1 -

20 points identified by Point Identification (PT ID) and time in Hours (HH), Minutes (MM), and Seconds (SS) in Column 1

Row 2 -

20 values of Point Identification (PT ID) and time in Hours (HH), Minutes (MM), and Seconds (SS)

g. Hourly logs - periodic (once per hour) on preprinted paper consisting of 14 logs with approximately 20 points per log on major systems of the plant including radiation and environmental data.

1.2.2.5 Capability for retention of data, information and physical evidence (both hardware and software).

Response

Data is retained on hardcopy reports, logs on magnetic tapes taken once per day by computer technicians (system saves), on disc storage and large core storage memory. Instantaneous values are located on tables in the computer and transferred to history areas on bulk storage devices for later recall. Weekly system saves to off-line disc storage provide updates of the information. In the event that a hardware failure ever occurred, the return to service would only be at the most one week by disc or 24 hours by tape from the present time. Additionally, special Nuclear Steam Supply System tables and arrays (denoted Data Classes 1-20) are transferred to hardcopy and magnetic tape frequently by Reactor Physics Technicians to a backup system called BUCLE. This provides Nuclear Steam Supply System calculations in the event of a failure of the hardware.

A backup paper tape system is also available but hardly ever used. Strip charts are saved and replaced as they are used.

1.2.2.6 Power Source(s) (e.g, Class IE, non-Classs IE, noninterruptable)

Response

Power sources are the same as those described in 1.2.1.6.

1.2.3 Other data an information provided to assess the cause of unscheduled reactor shutdowns.

Response

Other data and information available to assess the cause of unscheduled reactor shutdowns include operator logs, surveillance test data sheets, seismic recording equipment, operator interviews and occurrence reports. Minutes of review committee discussions on the events are also available.

1.2.4 Schedule for any planned changes to existing data and information capability.

Response

Since the data described in Section 1.2 is presently available and in use, with the exception of the previously noted Analog Trend History (Monitoring System) which is 70 percent complete, no specific changes are intended. A system regeneration of the history software and patched changes is in progress and should be completed by October 1984. This normal software maintenance activity takes the operating system, scan log and alarm subsystem, video system and Nuclear Steam Supply System system software and reassembles it in line to reflect the present system with the patched in changes that have occured since its installation. Additional video graphics which includes the Safety Parameter Display System (SPDS) is being implemented on this system and Piping and Instrumentation (P&I) Diagram on color displays is in progress.

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 nonnuclear 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.

2.1 Response

Current Equipment Classification and Vendor Interface (Reactor Trip System Components)

The equipment classification list (Q-List) used at Nine Mile Point Unit 1 has been preliminarily reviewed. The review included verification that components in the Reactor Protection System (RPS), are presently classified safety related except for the Reactor Protection System Motor Generator Sets. The Reactor Protection System Motor Generator Sets are currently classified as non-safety related and the remaining Reactor Protection System components are protected from malfunctions such as over-voltage, under-voltage, and under-frequency conditions in the Motor Generator Sets by electrical protection assemblies that are classified safety related. The safety related classification of other systems, such as Reactor Vessel Instrumentation System, Neutron Monitoring System, and Control Rod Drive System which comprise part of the "Reactor Trip Function", had also been reviewed. For these other systems, all or part of the systems are classified as safety related indicating that those portions of the systems which are associated with the "Reactor Trip Function" are properly classified. Additional investigation must be completed prior to any positive determination that all "Reactor Trip Function" components are classified safety related.

'2.1 Response (cont'd)

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.

These documents generally do not contain statements that a component is not safety related. Consequently, operators and maintenance personnel will not be misled by non-conservative mistakes in documents, procedures and information handling systems. The work request form 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 Supervisor checks the classification, again using the equipment classification list (Q-List).

Considering that classification information is minimized in the plant documentation, and that preliminary review of the equipment classification list (Q-List) has indicated proper classification of reactor trip components, reasonable assurance is provided that non-safety related work practices will not be conducted on safety-related equipment.

However, maintenance procedures will be reviewed to assure that any classification information is correct. This review will be completed in conjunction with the review of Section 3.1.1.

Nine Mile Point Unit 1 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 Operations Assessment Program, which receives, reviews and acts on applicable information from the reactor trip system supplier for Nine Mile Point. The information consists of General Electric Service Information Letters (SILs), Service Advice Letters, Technical Information Letters (TILs) and Product Experience Reports (PERs). Together with our current participation in the General Electric Operations Engineer Program, a high level of communication, feedback and equipment performance improvement is achieved. One example is an improved method of calibrating the main steam and offgas radiation monitors. This procedure was developed at Nine Mile Point Unit 1, fed back to General Electric and published as SIL 369.

2.1 Response (cont'd)

A preliminary review of Service Information Letters (SILs) listed in General Electric's Service Information Letters (SIL) status report has indicated that most Service Information Letters (SILs) listed in the "No Feedback" category were "For Information Only". Others in the "No Feedback" category were revisions or supplements to Service Information Letters (SILs) that Nine Mile Point Unit 1 had fully or partially implemented the previous issues. The preliminary review revealed that there are few, if any, outstanding Service Information Letters (SILs) involving the reactor trip system. Any that are identified will be addressed by May 1, 1984. 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 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. A preliminary review of the outstanding Notices, Significant Event Reports (SERs) and Significant Operating Experience Reports (SOERs) indicates few, if any, items concerning the reactor trip system. Any that are identified will be addressed by May 1, 1984.

The second part of the program is the Administrative Control of Technical Manuals. Nine Mile Point Unit 1 uses a "controlled copy" program to assure that the technical manuals available for use are the current revision. A program for periodic audit of the controlled copies is currently under development, and the first audit should be completed by May 1, 1984.

The third part of the program is the periodic review of maintenance procedures which includes a check to assure that the guidance of the "controlled copy" technical manual is referenced or incorporated in the procedure.

Planned and Potential Changes to Equipment Classification and Vendor Interface (Reactor Trip System)

Niagara Mohawk is participating in a Boiling Water Reactor Owners Group Committee which was specifically formed to address Generic Letter 83-28 related work. The Committee divided item 2.1 of the generic letter into several subtasks in order to provide generic responses to the maximum extent practical. These subtasks include:

Reactor Trip System Equipment List

The Committee has discussed several alternatives which have varying applicability to individual BWR utilities. Currently, Niagara Mohawk is evaluating these various options. A decision on the appropriate course of action is expected by February 1984. In the interim period, Niagara Mohawk's review of the Equipment Classification List (Q-List) discussed above has shown that the intent of the generic letter is being met.

Description of General Electric Information Programs

General Electric will provide a description to explain the information systems used to obtain operation and maintenance information from suppliers and how this information and similar information on General Electric manufactured items is provided to other utilities. This description is expected to be completed by February 1984.

Procedure for Safety Related Component Identification

In many instances, General Electric has performed safety classification of various components under their scope of supply.. The task will demonstrate General Electric's current method for classifying equipment. This task is scheduled to be completed by February 1984. The results will be evaluated for possible changes to the existing engineering procedures.

Niagara Mohawk currently has underway a program to upgrade its material management capability. Included in that program is a task to upgrade the details of the current Equipment Classification List (Q-List). For example, safety-related components will be broken down to the subcomponent level and then subcomponents classified as safety related or not. This task is expected to be completed by December 1984. This effort will provide a clearer definition to those components and subcomponents subject to the requirements of the Quality Assurance Program.

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:

2.2.1. 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.

Response

Control of safety related activities at Niagara Mohawk is accomplished by implementation of a Quality Assurance Program for safety-related systems, components and structures. Implementation of this program is achieved through administrative controls governing design, procurement, installation, testing, operation and maintenance by responsible Niagara Mohawk organizations. Responsibility for classification of equipment is vested in the Nuclear Engineering and Licensing Department of Niagara Mohawk. Existing administrative controls related to this function include:

Engineering Procedure 020 - Procedure for Classification of Materials and Components and for Use of the Q-List

Engineering Procedure 028 - Preparation and Control of the Q-Lists (EP 020; Appendix A; B; C) and the Excluded Equipment List (An Attachment to EP 240) and Revisions Thereto

Engineering Procedure 190 - Determination of 10CFR50; Appendix B Quality Requirements for Components or Services

The Equipment Classification List (Q-List) identifies those systems, components and structures to which the Quality Assurance Program applies. Activities associated with these systems, components and structures are required to be accomplished in accordance with the Quality Assurance Program.

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* Safety-related structures, systems, and components are those that are relied upon to remain functional during and following design basis events to ensure: (1) the integrity of the reactor coolant boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10CFR Part 100.

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

Response

General criteria relative to classifying components as safety related within safety related systems is contained in Engineering Procedure EP 020. This is accomplished by reference to 10CFR50, Appendix B and Nuclear Regulatory Commission Regulatory Guide 1.26, Rev. 3. In addition, Engineering Procedure EP 190 provides a vehicle for various departments within Niagara Mohawk to request an Appendix B determination as to the safety related classifications of components and services. When appropriate, the Equipment Classification List (Q-List) is updated to reflect Appendix B determinations.

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

Response

The current Equipment Classification List (Q-List) is provided in Engineering Procedure 020 as Appendices. Appendix A provides safety classifications for electrical components, Appendix B mechanical components and Appendix C for structures, components of structures and selected structural type equipment. Responsibility for preparation and control of the Equipment Classification List (Q-List) as well as review and approval are specified in Engineering Procedure 028. In addition, revisions to the Equipment Classification List (Q-List) are governed by Engineering Procedure 028.

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 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 by the Site Operations Review Committee. The final administrative control before work occurs is the requirement that the Shift Supervisor grant permission. 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 1 to have checks and balances in 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

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 plan 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.

2.2.1.5 Response

Attributes such as design verification and qualification testing associated with procurement of safety related components are addressed in appropriate engineering procedures. Engineering Procedure EP 90 Control of Design and Design Verification provides a method for design control and design verification. In addition, Engineering Procedure EP 100 Control of Procurement Activities identifies procedural provisions to ensure that the applicable regulatory requirements, design bases and quality requirements are met when obtaining material and services.

As part of our continuing electrical equipment environmental qualification program appropriate engineering procedures are being reviewed for possible revisions to more specifically address environmental qualifications of safety related electrical equipment.

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").

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.

2.2.2 For vendor interface, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information for safety-related components is complete, current and controlled throughout the life of their plants, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of safety-related equipment should be contacted and an interface established. Where vendors cannot be identified, have gone out of business, or will not supply 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, to assure reliability commensurate with its safety function (GDC-1). The program shall be closely coupled with action 2.2.1 above (equipment qualification). The 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 acknowledgment for receipt of technical mailings. It shall also define the interface and division of responsibilities among the licensee and the nuclear and nonnuclear divisions of their vendors that provide service on safety-related equipment to assure that requisite control of and applicable instructions for maintenance work on safety-related equipment are provided.

Response

Niagara Mohawk is actively participating in a Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, item 2.2.2. It is expected that this Committee will present its recommendations to the participating utilities in January 1984. Niagara Mohawk will then review these recommendations and will submit the Nine Mile Point Unit 1 status, plan and schedule by February 1984.

3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

Position

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.

Response

Administrative controls are in place to assure that post maintenance testing is completed. This control is achieved through statements in the maintenance procedures directing that a test be conducted. In some cases, the maintenance procedures contain the test. Maintenance and preventative maintenance procedures will be reviewed to assure that the correct post maintenance testing is specified.

A review of the Instrument and Control Department procedures will be completed by May 1984.

A review of the Maintenance Department procedures will be completed by November 1984.

A review of Technical Specifications will be completed by February 1984.

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.

Response

Nine Mile Point Unit 1 has performed a preliminary review of the status of Service Information Letters (SILs). Service Information Letters (SILs) on reactor trip system components will be re-reviewed for test guidance and incorporated into procedures as appropriate by May 1984. This re-review will be part of the Service Information Letter (SIL) review of Section 2.1.

3.1.3 Licensees and applicants shall identify, 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.)

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3.1.3 Response

The review of post-maintenance testing requirements in Technical Specifications will be done in conjunction with item 3.1.1 above.

3.2 POST-MAINTENANCE TESTING (ALL OTHER SAFETY-RELATED COMPONENTS)

Position

The following actions are applicable to post-maintenance testing:

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

3.2.1 Response

Post-maintenance testing of all other safety related equipment is administratively handled in the same manner as described in 3.1.1.

A review of the Instrument and Control Department procedures will be completed by November 1984.

A review of the Maintenance Department procedures will be completed by November 1984.

A review of Technical Specifications will be completed by February 1984.

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.

Response

Nine Mile Point Unit 1 has obtained from General Electric a listing of the status of Service Information Letters (SILs). Service Information Letters (SILs) on other safety related systems will be reviewed for test guidance and incorporated into procedures as appropriate by November 1985. This review will be part of the SIL review of Section 2.2.2.

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.

3.2.3 Response

The review of post-maintenance testing requirements in Technical Specifications will be done in conjunction with item 3.2.1 above.

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 (see Action 4.3 above) and CE plants; the circuitry used for power interruption with the silicon controlled rectifiers on B&W plants (see Action 4.4 above); and the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.

Response

Niagara Mohawk is particpating in a Boiling Water Reactor Owners Group to address this item. Considering the similarity of the reactor trip system of virtually all boiling water reactors, an Owners Group is the most effective means of developing an appropriate response to this item.

General Electric will evaluate the adequacy of the current specifications for functional testing of the resistor trip systems and will address functional testing of the scram pilot valves and backup scram valves. It is anticipated that the results of this evaluation will be available in February 1984.

4.5.2 Plants not currently designed to permit periodic on-line testing shall justify not making modifications to permit such testing. Alternatives 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.

Response

On-line functional testing of the reactor trip system, including independent testing of diverse trip features, is performed regularly at Nine Mile Point Unit 1. The surveillances, 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 "make up", actuating the entire trip channel up to and including one of the two scram pilot valves on every control rod's scram inlet and discharge valves and one of two backup scram valves. Periodic scram testing is performed during each cycle.

In addition, Nine Mile Point Unit 1 records the scram times of 30 control rods upon every reactor scram.

This scram time testing demonstrates the action of the pilot scram valves and scram inlet and discharge valves.

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. uncertainties in common mode failure rates
3. reduced redundancy during testing
4. operator errors during testing
5. component "wear-out" caused by the testing

Response

Niagara Mohawk is particpating in another committee of the Boiling Water Reactor Owners Group formed to address potential Technical Specification improvements. This Committee is expected to finalize their program by February 1984. The original committee objectives addressed the attributes discussed above.