NSAC/6

November 1980

MASTER

Workshop on Data-Acquisition and [#]Display Systems: Directions After TMI

Prepared by
Technology for Energy Corporation

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Workshop on Data-Acquisition and
-Display Systems:
Directions After TMI

NSAC-6

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NSAC PERSPECTIVE

This workshop may be characterized as "a beginning" in contributing to an orderly change of industry practices. Change requires a synthesis of thought, the bringing together of ideas, and, most importantly, the development of a consensus. The workshop provided the beginning for changes in data acquisition and display, which will be made in response to the Three Mile Island (TMI) experience.

The concrete results of the workshop are documented in some detail by this report. Included are (1) a fundamental parameter set for PWR and BWR designs, (2) a procedure for parameter set validation, (3) criteria for selecting parameters to be made available in the Technical Support Center, (4) criteria for the function and design of a safety console, (5) comparison of digital versus analog approaches to control room upgrading, and (6) functional considerations relative to the nuclear data link.

It is clear in retrospect that post-TMI data acquisition and display requirements represent a substantial industry concern. However, views expressed by workshop participants suggest that the issue is multifaceted. For instance, there is concern that early commitment to data acquisition and display improvements will be superseded by later requirements. There is concern that system improvements without careful validation will lead to display panels or data handling systems of no particular use to plant operators. There is concern that present implementation schedules will lead to a piecemeal, backfit approach which will not be conducive to integrated systems applications. Finally, there is concern that the scramble to put near-term plant modifications into place will subordinate improvements with greater safety significance, which are implementable only in the longer term.

The current industry situation is fragmented at the local level, quite possibly as a result of these fundamental concerns. It is evident that neither proprietary nor parochial considerations are presently overriding factors. Rather, a workable mechanism for a continuing technical dialogue between industry groups, which can be carried out in conjunction with a negotiating effort with the Nuclear Regulatory Commission (NRC), seems to be lacking. Both the dialogue and the negotiations must be undertaken on a fairly generic (industry-wide) basis in order to be successful.

The workshop provided a start for an industry-wide technical dialogue. The discussions led to a differentiation between short-term and longer-term objectives. In the near term, high priority must be given to the selection of a fundamental set of plant parameters which could be applied to multiple data acquisition and display needs. It was recognized that a common set of parameters should be identified, which could provide the basic data set for a safety panel in the control room as well as basic information for the on-site Technical Support Center, the Emergency Operation Center, and the Nuclear Data Link. This parameter set should be standardized in the short term for PWR and BWR systems, and it should provide an "acceptable" basis for near-term responses to TMI data acquisition and display requirements. Parameter set selection and validation should be a priority item.

In the longer term, it is clear that gradual transition towards a flexible system for handling and displaying plant data is a sound and perhaps inevitable course of action. A digital, software-based system would presumably be used in this capacity. It is recognized that the logistics and general availability of technical personnel may rule out this option in the near term for many operating plants. Nevertheless, the technical dialogue should be continued in order to establish guidelines and design criteria for these more advanced systems. This will assure that the new technology will be constructively applied, rather than merely being adopted for its own sake.

The conclusions drawn and general recommendations herein should be interpreted with knowledge that the workshop was organized on an "ad hoc" basis, with input from the Atomic Industrial Forum Subcommittees on Control Room Considerations and Emergency Response Planning.* As such, no decision-making authority was conferred or assumed and no actions taken for, or expressed in behalf of, the nuclear industry.

*AIF Policy Committee on Follow Up to the Three Mile Island Accident

FOREWORD

This report synthesizes the comments and thinking of more than 30 individuals. These individuals come from varied backgrounds and represent several viewpoints. A draft of this report was provided to the participants immediately following the conference for their review and comments. All comments received either in writing or by telephone were incorporated; however, NSAC recognizes that the potential for misinterpretation may still exist in some areas. Hence, neither this report nor any section thereof should be contrived as representing the views of any particular individual or organization. Rather this report attempts to raise the significant issues and synthesize the views of all participants.

ABSTRACT

The accident at Three Mile Island Unit-2 raised questions as to the adequacy of data acquisition and display systems in commercial nuclear power plants. A series of recommendations have developed from the various groups that have analyzed the accident in order to improve the operator's overview of the plant safety conditions and to facilitate information transfer to technical support centers in emergency situations. This report is the result of an NSAC-sponsored workshop, where the various recommendations and emerging regulatory requirements were reviewed in an attempt to provide an integrated basis for their implementation.

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WORKSHOP ON DATA ACQUISITION
AND DISPLAY SYSTEMS--DIRECTIONS AFTER TMI-2

1. INTRODUCTION

The Nuclear Safety Analysis Center (NSAC) sponsored a workshop March 18-19, 1980 to undertake a comprehensive review of the various data acquisition and display system requirements which have developed as an outgrowth of the TMI-2 accident. There were 37 participants from the AIF Control Room Subcommittee, EPRI, Near Term Operating License (NTOL) Plants and other utilities, NSAC, NSSS Vendors, Quadrex, and Technology for Energy Corporation.

Since the TMI-2 accident, a number of studies have pointed to the need for improved data acquisition and display systems to facilitate the control and mitigation of potential nuclear power plant accidents. A series of recommendations have been made by NRC, the AIF Policy Committee on Follow-up to the Three Mile Island Accident, the Kemeny and Rogovin Commissions, and industry groups studying the TMI accident. These recommendations have ranged from control room design review and upgrading of existing display systems, to the implementation of entirely new instrumentation panels and real time transmission of data from control rooms to remote locations. Many of these recommendations are in the process of being codified as regulatory requirements with specific acceptance criteria and dates for implementation.

There is increasing belief that the functions and data requirements for many of the new systems are interrelated. However, the extent of this interrelationship and how these systems should be interfaced with existing control room equipment has not been established. Furthermore, little detailed consideration has been given to the duties of control room and emergency support personnel and the particular information to be supplied to them. In addition, the NRC implementation

schedules may not allow a coordinated approach in meeting new requirements. Furthermore, individual utilities are making commitments to obtain operating licenses which may establish industry practices simply on the basis of precedence. These problems are contributing to a piecemeal resolution of these issues by the industry and NRC.

1.1 Purpose

The purpose of the NSAC workshop was to bring together cognizant individuals from utilities and vendors to examine the basis for a more consistent understanding and approach in meeting TMI-related requirements.

The objectives of the workshop were to:

1. Define the issues involved and the validation required for establishing the fundamental parameters which define nuclear power plant safety status. As discussed in the March 14, 1980 letter from the AIF Policy Committee to the Three Mile Island Accident, "these parameters might be integrated into a safety panel design or utilized in some other control room evolution developed through design reviews now underway."
(Appendix A)
2. Formulate a framework for integrating the operator aids proposed as a result of TMI-2 and to outline functional specifications for accomplishing this framework.
3. Define functional specifications and address critical issues involved in establishing the following data acquisition, display and monitoring systems being proposed to meet TMI-related requirements: (a) Plant Safety Parameter Display Console (Safety Console)*, (b) Safety System Status Monitoring (Regulation Guide 1.47), (c) Technical Support Center

*This system is also referred to as the Safety System Monitor and the Safety Console. Safety Console is the term used throughout this report.

(TSC), and (d) Nuclear Data Link (NDL).

A detailed summary of proposed TMI-related requirements that bear upon or directly relate to data acquisition and display is contained in Appendix B.

In addition to these formal objectives, the workshop also had a number of important informal objectives: (1) To stimulate communication between a broad-based group of representatives from the nuclear industry; (2) To establish a common basis for understanding the evolving NRC TMI-related requirements; and (3) To define a basis for establishing a coordinated technical dialogue between NRC and the industry. The goal was to identify a mechanism by which the two groups could work together to understand the need for and to define sound technical and functional specifications for the new systems required.

1.2 Organization

Four working groups were organized. Group A was asked to formulate a framework for integrating operator aids and to discuss: (1) the control room design review; (2) the needs of the shift technical advisor; (3) Regulatory Guide 1.97; and (4) alternative system architectures including the process computer. Group B was asked to develop a scope statement defining the objectives, the design criteria and a list of major issues associated with the Safety Console and the Technical Support Center. In addition, Group B was asked to address issues raised by Regulatory Guide 1.47 concerning safety system status monitoring. Group C was asked to develop a set of guidelines to be followed in defining the Technical Support Center's data acquisition requirements and discuss the data acquisition issues involved if a Nuclear Data Link is established. Group D was asked to develop a framework for establishing the fundamental parameter set required to monitor the status of nuclear power plant safety. The agenda for the workshop and the list of participants are contained in Appendix C.

The workshop began by hearing reports about industry efforts to date to meet impending NRC requirements from the Near Term Operating License (NTOL) Plants, the Westinghouse Owners Group Instrument Systems Ad Hoc Committee and General Electric Owner's Group Control Room Subcommittee. These efforts are summarized in Section 2. In addition, there was a report on the Disturbance Analysis and Surveillance Systems programs (Appendix H). Sections 3 - 8 report the results of the working groups, their specific recommendations, and the general discussion which followed the working group sessions. Section 9 presents the general recommendations of the workshop.

2. EFFORTS OF INDUSTRY TO DATE TO MEET NRC REQUIREMENTS

2.1 Near Term Operating License Plants (NTOL)

Representatives from four NTOL Plants -- Diablo Canyon, Salem, Sequoyah, and North Anna -- reported on their current plans and commitments to meet the TMI-related NRC requirements. Table 2.1 summarizes NTOL efforts to date.

2.2 Owner's Groups

Representatives of the Westinghouse Owner's Group Instrument Systems Ad Hoc Committee and General Electric's Owner's Group Control Room Subcommittee reported on the systems they are developing to meet new NRC requirements.

2.2.1 Westinghouse Owner's Group Ad Hoc Instrument Systems Committee

The Westinghouse Owner's Group Instrument Systems Ad Hoc Committee presented a document, "Technical Support Center and Related Control Room Instrumentation System Implementation Guideline." The document describes a set of functional requirements which it is believed meets the intent of NUREG-0578 for a Technical Support Center (TSC) and can be implemented before January 1, 1981 (Phase A). The document also identifies and tabulates additional requirements for the TSC and related control room instrumentation which is projected will require implementation on a two to five year schedule (Phase B). A schematic of the proposed system is contained in Appendix D.

2.2.1.1. Phase A. The document defines those parameters displayed in the control room which should be monitored in the TSC to provide historical and real time plant data and also defines the general equipment requirements. The Ad Hoc Committee recommended that: (1) TSC instrumentation systems not be classified 1E; (2) TSC instrumentation need not be seismically qualified; and (3) plant

TABLE 2.1
INITIAL EFFORTS OF NEAR TERM OPERATING LICENSE (NTOL) PLANTS TO MEET NEW NRC REQUIREMENTS
AS OF MARCH 1980

	DIABLO CANYON	SALEM	SEQUOYAH	NORTH ANNA
Control Room and Design Review	P-250 computer. Have developed and are in the process of installing Data Link System (DLS). 95% of software completed; delivery of off-shelf items takes 45 days; and installation can be completed in 3 months for \$200,000. Paint and tape changes are anticipated. Do not expect extensive rearrangement of display instruments or controls. No official commitment to date.	Agreed to low power testing safety review group of people. Submitted formal company control room review which called for changes such as high-lighting tape, switch guards, and some minor instrumentation location. Essex Corporation and NRC presently on site conducting 5 day review.	Several minor changes as a result of Essex report namely (1) rail guards on control board, (2) fluorescent tape on board separating Units 1 and 2. room with P-250 computer. Have installed carpeting.	P-250 computer. Following meetings with NRC, have identified all the concerns with regard to human factors and man interface. A few significant changes are planned but most changes are minor. A one-year initial modifications which may result in major modifications. Could possibly involve replacing existing process computer system.
Display Console	No commitment to date. Looking at various ways to display data possibly some sort of colorgraphic display.	Following industry efforts, no commitment made to date.	No commitment to date. Have some overall ideas as part of TSC but nothing firm now.	At present, committed to using colorgraphics. Results of one year study may affect this.
Reg. Guide 1.47	Administrative System	Have computer-assisted administrative system for valve status. Mimic display is provided for reactor protection and emergency safeguard status on back panel.	Have always had a system CRT display that gives status of valves, motors, electric boards, etc. Some problems with software. They are expanding number of points. Have put a few more items on present system.	Plan to accommodate with requirements but no commitment made to date. Presently formulating what will be required to meet 1.47.
Technical Support Center	TSC can be linked into DLS by adding interactive terminal, type RS-232 interface. Data link with existing process computer system to new data handling computer system in TSC, featuring direct tie-in with batch processing centers and remote interactive terminals.	Currently have room on 3rd floor of new services building with 15-20 telephone lines, slave control room CRT, trend typewriter for selected parameters, and ability to call up any parameter available on process computer. Still evaluating method for satisfactory long-term instrumentation requirements. HVAC and shielding will be upgraded.	TSC room with a closed circuit TV monitor connected to a hand-held camera. No commitment to do more now. Expect to do more before getting full power license. Looking to build separate building to house TSC.	Committed to having temporary center with sufficient capability now and additional building in the future with expanded capability.
Operational Support Center	OSC can be linked into DLS.	Have established area between control rooms and have prepared procedures.	General assembly area. Do not feel they need much more than that.	Have designated location and committed to procedures to address role of OSC.

TABLE 2.1
INITIAL EFFORTS OF NEAR TERM OPERATING LICENSE (NTOL) PLANTS TO MEET NEW NRC REQUIREMENTS
AS OF MARCH 1980
(Continued)

	DIABLO CANYON	SALEM	SEQUOYAH	NORTH ANNA
Emergency Operations	EOC can be linked into DLS. Trailer being built next to the Sheriff's Office with another terminal linked to DLS.	Have established Emergency Control Center near the site and feel it is better for it to be off-site rather than adjacent to the property line. The majority of engineering personnel would report to the Corporate office.	Emergency Control Center located in Chattanooga. Press facilities near airport. Emergency Operations Center (EOC) near the plant still undecided.	No commitment to date.
Nuclear Data Link (NDL)	NRC can hook up to DLS. Planning to be able to transmit any or all plant computer data with interactive capability; also transmit to state, county sheriff, I&E regional headquarters, and home office.	No commitment made to date.	Have only discussed in general with NRC.	Plans to accommodate with requirements but no commitment made to date.
Status of License	Hearing scheduled to be held in June on security plan. Following hearing hope to know status of license.	Expect receipt in early April of 5% operating license SER being drafted by NRC.	Present license limits them to 5% power. License is conditioned to comply with Reg. Guide 1.97 by 7/1/82.	Meetings with NRC Commissioners have been completed. They are awaiting final decision.
Reg. Guide 1.97	No commitments beyond NUREG-0578.	Have provided comments on draft 1.97. No commitment made to date.	Participated with AIF committee in commenting on draft 1.97 to NRC. NRC has required compliance with 1.97 in order to obtain license.	Have submitted comments on draft 1.97. Awaiting final 1.97 which they plan to implement.
Other NRC Requirements	None	Have Shift Technical Advisor. Have 30-week in-house training. Also, Westinghouse is providing a 13-week training course. Have had meetings with NRC on Reg. Guides 1.97 and 1.47.	Expanding list of surveillance requirements.	Human Factors: Know of no requirements to date.
Problems	Environmental Monitor Data System being designed. DLS transmits raw data from control room; not sure what various centers will do with this raw data.		Reg. Guide 1.97 qualification levels for diagnostic instrumentation. Environmental qualification - NRC not sure of technical specifications. RID's not qualified to radiation level. Shift Technical Advisor role and function a problem.	Difficulty in establishing data link with P-250. Time involved to institute plant to meet all NRC requirements. Trying to develop overall company plan to cover these requirements.

instrumentation systems which are Class 1E and which provide inputs to the TSC should be isolated so as not to degrade any safety functions.

2.2.1.2 Phase B. The document also identifies functional requirements, the associated system hardware and software requirements, and a suggested implementation schedule for computer-based Technical Support Center and control room instrumentation systems beyond the NUREG-0578 requirements addressed in Phase A. Regulatory Guide 1.97 may require plants to install additional sensors and circuits which include Class 1E systems. Therefore, the Westinghouse Owners Group Ad Hoc Committee recommended that the installed system have the capability to accommodate the Class 1E requirements.

The committee plans to present this system approach to NRC April 2, 1980.

2.2.2 Preliminary Recommendations of the General Electric Owner's Group Control Room Subcommittee

The General Electric Owner's Group Control Room Subcommittee was formed in January 1980. Their preliminary recommendation for the functional and equipment requirements for a Nuclear Data Link (NDL) system is to use the existing plant computer. Data can be transferred to a dedicated system from which a "Data Link System" and other similar requirements could be accommodated. (This system is modeled after the PG&E approach.) They described the Data Link System as (1) "using off-the-shelf components which have a delivery schedule of 45 days, (2) having a schedule which can be completed in 3 months, and (3) having software (at PG&E) which is approximately 95% complete." The interface with the data link and other acquisition and display systems has not yet been defined.

The Data Link System, as discussed by the GE Owners Group Subcommittee, is capable of transmitting data from the Control Room to the Technical Support Center and the Nuclear Data Link. It is not yet defined how the various centers would use the data. Block diagrams of the system are in Appendix E.

The Subcommittee's future plans are to: (1) Present the system to the NRC for review by Sandia Labs, which is developing the NRC Operations Center data collection system; (2) Develop a generic BWR approach to the Data Link System; and (3) Continue efforts to develop control room modifications for the Safety Console and Technical Support Center.

2.3 AIF Policy Committee on Follow-up to the Three Mile Island Accident

The work of the AIF Policy Committee on Follow-up to the Three Mile Island Accident was briefly discussed at the workshop. This committee has defined an emergency response organization which is a "generic" or typical plan which could be used as a guide by each utility/operator of a nuclear power station.

The basic response plan defines a framework of people and facilities which should be employed in an emergency. It recognizes that a utility may require resources beyond its own in-house capabilities to minimize the impact of a nuclear accident on the public. Figure F-1 in Appendix F describes a typical emergency response organization to perform the variety of functions and activities necessary during an emergency. Figure F-2 in Appendix F shows typical facilities and communications arrangements to support the requirements of the response organization.

2.4 General Session Discussion on Initial Industry Efforts to Meet TMI-Related Requirements

2.4.1 Standardization

Standardization was listed as a critical issue in industry activities to meet the TMI requirements. Each of the Near Term Operating License (NTOL) plants is responding to the requirements with a different technical approach. This may be the result of utilities trying to "second guess" evolving regulatory requirements. However, there is also apparent inconsistency in the way in which different NRC reviewers are interpreting and applying the developing requirements to each plant.

One NTOL plant is upgrading their control room by applying fluorescent tape to the control board and another is committed to buying totally new systems. In the latter instance, it appears that the utility strategy is to implement a flexible data handling system and await the development of "hard" regulatory requirements.

The NTOLs believe they are doing the best they can under the current circumstances and face potential economic losses if they delay further. The general concern is that early commitment under these pressures may set industry precedents which may produce inadequate or poorly developed or uneconomic generic responses to safety issues.

Standardization is needed for the more complicated systems and concepts being proposed. NRC and the industry need a mutual understanding on the intent and function of the new requirements in very specific terms in order to achieve a more consistent approach in safety improvements.

2.4.2 Schedule

Concern was expressed that the schedule for implementation does not allow time to set priorities and work out technical problems. Care must be used to avoid over-commitments to new concepts. Research and development may be required in some areas prior to implementation.

2.4.3 Near-term/long-term solutions

The merit of a near-term/long-term phased approach needs to be discussed. Near-term improvements may be good for only a few years. For instance, no long-term decision as yet has been made as to what type of system is needed in the Technical Support Center i.e., digital or analog, or how to put it together. If a utility commits to a highly constrained, fixed system in the short-term, it may be forced to replace it later.

3. INTEGRATING NEW REQUIREMENTS WITH EXISTING SYSTEMS

The industry and NRC are considering data acquisition and display systems to provide tractable information in the control room and in several other places in case an accident occurs. In the control room, NRC has focused on: control room design reviews; a "Safety Console" to provide an overview of plant safety status; and more in-depth implementation of Regulatory Guide 1.47 (Safety System Status Monitoring System). Outside the control room, but presumably within the plant perimeter, the need for a Technical Support Center has been identified with a Recovery Center and an Emergency Control Center near site. NRC and others, including state and local governments, utility headquarters, architect-engineers, and vendors are identifying their needs for a data link with the control room.

These needs for communications flow and specific information in several facilities raise two fundamental and difficult questions:

1. What are the functions and hence the data needs of each of these facilities?
2. How are new requirements for the data acquisition and display systems serving the new facilities to be integrated with existing systems?

There are many unresolved issues associated with these questions for the several facilities and systems under consideration. These issues are discussed in some detail in Sections 5 through 9 of this report. Many believe that the industry should establish a common minimum set of parameters for safety assurance and supervisory overview which could be integrated into a Safety Console and perhaps could serve as the basic data set for many of the other facilities and com-

munication links being identified. This kind of fundamental parameter set is discussed in Section 4.

Group A focused on the second question stated above, i.e., how the data acquisition and display systems are to be integrated with existing systems. The scope and objectives of the group and the issues to be addressed in answering these questions are discussed in this section.

3.1 Scope and Objective

Group A's scope was to formulate a framework for integrating the operator aids proposed as a result of TMI-related requirements and to outline functional specifications for accomplishing the framework.

3.2 Background Review

The following background materials were reviewed by Group A:

1. Regulatory Guide 1.97 and Regulatory Guide 1.47
2. NUREG 0660 regarding NRC Operating Plant Control Room Review
3. NUREG 0578, 0585, and 0660 (Draft 3)

3.3 Issues to be Addressed for Integrating the Application of the Various Operating Support Systems

1. Should the Safety Console, the Technical Support Center, Regulatory Guide 1.47, and the Nuclear Data Link be digital or hardwired systems? Table 3.1 lists the advantages and disadvantages of the two systems.
2. The Nuclear Data Link requires a computer; therefore, should other functions be considered for this equipment, e.g., TSC expandability?
3. Long term objectives for these systems should assume a digital system.
4. Should the economical short-term implementation be simple digital systems which can be expanded to provide the long-term objective?
5. For digital, computer-based systems, the following issues need to be addressed:
 - a. Software Considerations
 - (1) Application software validation and verification methods (standard)

Table 3.1

Digital vs. Hardwired Systems for the Safety Console,
the Technical Support Center and the
Nuclear Data Link

<u>Hardwired Advantages</u>	<u>Hardwired Disadvantages</u>
Instantaneous, continuous historical record	Requires more control room space
Conventional to operating "mind-set"	Cabeling and equipment installation
Qualification nearer completion	Permanent
Industry is comfortable and familiar with it	Historical data manipulation not available
I&C repair is well understood and utility addressed	Drift and calibration problems
Equipment does not undergo same obsolescence.	Strip chart recorder data analysis extremely difficult
	Difficult and costly to transmit long distances
	Difficult and costly to expand functional capability
<u>Digital Advantages</u>	<u>Digital Disadvantages</u>
More functional and flexible	Software is expensive
Hardware economical	Software validation and verification
Reduced impact on plant	New capabilities for utility personnel
One set of hardware will accomplish many functions	Obvious departure in control room display (CRT)
Impervious to data processing noise (higher precision)	High technological obsolescence rate (no spare parts)

- (2) Operating system software validation, verification, and modification
 - (3) Software reliability goals for these functions
 - (4) Software design requirements for upgrade and functional expandability
- b. I/O, Human Factors Considerations
 - (1) Human factors integration of displays (CRT's)
 - (2) Cognitive requirements for operator decision making processes. (These cognitive requirements must underlie the design and evaluation of the CRT display.)
 - (3) "Proper" implementation of displays: to whom, where, how often?
 - (4) Inclusion of human factors technology into procedures, training, and operator "mind-set"
 - (5) Controlling access to and from system data base. (Should data links be one-way, free-access to all data bases? Should they be interactive?)
 - (6) Scanning frequency requirements
- c. Architecture/Compatibility (Hardware) Considerations
 - (1) Reliability questions (acceptable architecture)
 - (2) Hardware reliability goals for these functions
 - (3) Scanning frequency requirements
 - (4) Hardware design requirements for upgrade and functional expandability
- 6. If available and capable, in-plant process computer offers easy basis for upgrading and feeding these systems.
- 7. Control Room Reviews and Upgrades
 - 1. Impact of upgrade on operator "mind set"
 - 2. Impact of upgrade on training, procedures and crew role definitions
 - 3. Impact on operator requirements, both physical and mental
 - 4. Method of upgrade implementation; e.g., should no upgrade take place until operating crews are retrained?
- 8. Location of Safety Status Panel and to whom and for what purpose will information be displayed?

3.4 Group A Conclusions on Integrating New Requirements with Existing Systems

1. Equipment. The Regulatory Guides have not specified what equipment and systems are required. The attendant hardware issues must be studied in depth. A preliminary review of equipment issues led to the following general statements by Group A.
 - a. Regulatory Guide 1.97 seems to require hardwired systems in the control room.
 - b. Regulatory Guide 1.97 may define the encompassing data set for all operating support systems.
 - c. Non-IE computer systems appear to be the most efficient for implementation.
 - d. Digital systems appear to be the best method of implementing these functions in the long term.
 - e. The existing process computer (if available and capable) seems to be the most efficient provider of a data base.
 - f. The best short-term implementation appears to be with small digital systems which can be readily expanded in a logical and economical fashion to achieve the anticipated long-term functions.
2. Data Base Access. Data base access and use needs to be controlled. An indepth review is needed of data flow, format, and handling.
3. Implementation. Implementation of any operator aid system needs to have: demonstrated (tested) operator performance improvement; and careful inclusion in training and plant procedures.

4. FUNDAMENTAL PARAMETER SET

Many believe that the industry should develop a common minimum set of parameters for safety assurance and supervisory overview. Such data could be integrated into a Safety Console or status monitor and perhaps serve as the basic data set for many other facilities and communication links being considered within the industry and by NRC. Group D focused on such a data set. The approach followed by the group was to: (1) define the scope; (2) consider previous work done to define a fundamental parameter set such as parameter lists or criteria from NSAC, ANS 4.5, Regulatory Guide 1.97, the AIF Policy Committee on Follow-up to the Three Mile Island Accident, and from informal NRC staff input; (3) select parameters; and (4) identify tests to validate the parameter set such as WASH 1400, and those identified as appropriate through either recent or historical events.

4.1 Scope and Objective

Group D's scope and objective was to establish the fundamental parameter set for the overview of nuclear power plant safety status.

4.2 Guiding Principles

There were three principles that served as a basis for the group's work: (1) maintaining barriers to radioactive release should be a primary accident/post accident operations objective; (2) the selected parameters should define the status of barriers to radioactive release; and (3) human factors theory on mental correlation should be considered in establishing an upper limit on the parameter set size.

4.3 Critical Issues

The issues that Group D thought were critical to consider in defining a fundamental parameter list were: (1) design criteria; (2) problems such as over reliance and false assurance; and (3) the acceptability of alternative designs.

4.3.1 Human Factors

Given the variables selected, how should they be combined to represent safety indicators? Human factors research has shown that, if given more than 9 indicators, the human mind cannot, in the short term, evaluate them. Furthermore, there is high variability in the operator's hierarchy of understanding. Once a transient begins to evolve, one person can only follow a few indications far into an accident.

4.3.2 Other Technical Issues

Other technical issues needing further attention are:

1. Parameters which define the status of the barriers to radioactive release.
2. How to best monitor core cooling.
3. Calibration range of the parameters. Should they be the same as the others in the control room?

4.4 Group D Conclusions on the Fundamental Parameter Set

4.4.1. Parameter Validation

The fundamental parameter set should be validated by consideration of the following items:

1. WASH 1400. Describe the classes of potential events whose probability warrants inclusion of certain process parameters.
2. FSAR Chapter 15 Events. Events analyzed in the plant safety analysis must be appropriately considered in the parameters selected for the safety status monitor.
3. Plant Specific. Events that are particular to a PWR or BWR, or that are

known to be characteristic of certain NSSS or Balance of Plant (BOP) design.

4. Mode of Operation. The proposed data set must be tested against known modes of operation, e.g., the six modes described in the standard technical specifications.

The purpose of parameter validation is to ensure that for the events addressed in the items above, the user (whether operator, TSC, or data link) can obtain a correct supervisory indication of the plant safety status.

In performing this verification, a "test" approach can be used; that is, once an event is identified, the critical parameters that are indicative of this event should appear on the Safety Console. Thus, an initial definition of parameters is not intended to be definitive, but should be expanded (or contracted) as additional operating experience is gained or as new classes of events are identified. In addition, other aspects of parameter definition, such as required ranges must be considered in the verification process.

4.4.2. Fundamental Parameter Set

Group D defined a preliminary parameter set (see Table 4.1) and compared it to those sets developed by TVA, NSAC, ANS 4.5, and selected NRC documents. It was recommended that the nuclear industry continue to participate in the further development and validation of this basic parameter set before it is finalized. Continued work should be undertaken to further develop and validate a parameter set as a priority activity.

4.5 General Session Discussion on the Fundamental Parameter Set

4.5.1 Parameter Selection and Validation

There was general concurrence with the objectives of AIF (Appendix A) to develop a minimum set of parameters for data acquisition and display systems.

Questions were raised about how the monitored variables are to be selected and how it is to be proven that they adequately represent the safety of the plant

Table 4.1
PWR FUNDAMENTAL PARAMETER LIST

List	TVA	R.G. 1.97 Cat. B&C	Westinghouse Owner's Group Subcommittee	ANS 4.5 Cat. B&C	Informal NRC Staff List	NSAC Staff List	NSAC Workshop PWR List	NSAC Workshop BWR List
Barrier								
Fuel Clad (Evidence of Inadequate core cooling or fuel failure)	THot TCold SG Level SG Pressure Containment RAD Monitor	THot TCold Containment RAD Monitor Core Exit T/C's CST Level RCS Flowrate SG RV Position RCS Radiation Monitor	THot (or core exit T/C's) TCold (or core exit T/C's) SG Level SG Pressure Containment RAD Monitor Aux FW Flow Condenser Air Ejector RAD Monitor	THot TCold RCS Radiation Monitor Neutron Flux	THot (or core exit T/C's) SG Level Containment RAD Monitor RHR Flow RCS Flowrate RCS Radiation Monitor Condenser Air Ejector RAD Monitor RHR Radiation Monitor	THot TCold SG Level SG Pressure Containment RAD Monitor Core Exit T/C's Aux FW Flow RCS Flowrate Main Facility Exhaust Monitor CR Position	THot TCold SG Level SG Pressure Containment RAD Monitor Core Cooling, Vessel Level or Exit T/C's	Rx coolant radiation Reactor Vessel water level
Reactor Pressure Boundary	Pressurizer Level RCS Pressure Containment Sump Level RWST Level	Pressurizer Level RCS Pressure Containment Sump Level SRV & PORV Position Boric Acid Charging Flow	Pressurizer Level RCS Pressure Containment Sump Level HPSI Flow	Pressurizer Level RCS Pressure Containment Sump Level Neutron Flux	Pressurizer Level RCS Pressure Containment Sump Level Drain Tank Level Boron Concentration Neutron Flux RCS Vessel Level Coolant Subcooling	Pressurizer Level RCS Pressure Containment Sump Level Drain Tank Level CVCS Tank Level Boron Concentration Neutron Flux Letdown Flow	Pressurizer Level RCS Pressure Containment Sump Level	BWR RCS Press- ure Drywell Sump level Reactor vessel water level
Containment	Containment Pressure Containment H ₂ Concen- tration	Containment Pressure Containment H ₂ Concen- tration Cont. Iso. Vlv. Position Heat Removal by Cont. Fan Coolers	Containment Pressure	Containment Pressure Containment H ₂ Concen- tration Cont. Iso. Vlv. Position	Containment Pressure Containment H ₂ Concen- tration	Containment Pressure Containment Temperature	Containment Pressure	Contain- ment Pressure H ₂ concen- tration RCS Press- ure Reactor Building Radiation

under yet undefined conditions. Human factor concerns were apparent; it was clear that mental correlation of more than 7 or 8 parameters is not feasible, but that several times more signals will be required to describe the safety state of the system. Therefore, grouping of parameters according to safety function will be essential.

It was concluded that tests of the parameters need to be run in operator training situations using a simulator.

5. SAFETY CONSOLE

NUREG 0660 draft, dated March 5, 1980, states that:

"NRR will require all licensees and applicants to install a safety parameter display system that will display to the operator a minimum set of parameters (safety state vector) which define the safety status of the plant. The system must have the capability of permitting the operator to display a full range of important plant parameters and data trends on demand. In addition, the system must provide indication to the operator that process limits are being approached or exceeded."

5.1 Data Acquisition and Display Systems -- The Plant Operators Viewpoint

An overview of data acquisition needs was provided by a representative of Duke Power/NSAC. The objective of the Safety Console concept should be to provide an aid to operations personnel during nuclear plant transients. In designing the console to meet this objective, two main issues are instrumentation usage improvements and the operator's approach during the various modes of operation.

First, the control room instrumentation must aid the operator in monitoring and responding to two kinds of transients--design or anticipated transients (start up/shut down/LOCA's) and complex transients (simultaneous Loss of Feed and LOCA). The instrumentation placement currently in the Control Room requires the operator to perform a process in order to acquire needed information. The operator must gather the applicable information from various locations and develop a directional trend. With this information a selection of available options is made to stabilize a transient. This process may be acceptable for the design transients. However, complex transients with accompanying conflicting or similar symptoms reduces the ability of the operator to effectively perform this process. Therefore, data acquisition and display systems should be more centra-

lized in a trended format to assist the operator in mitigating the effects of transients.

Second, the current approach to operation during the various transients needs to be augmented with a functional approach. The functional approach is derived from operating fundamentals (i.e. core covered, core cooled, containment etc.) applicable during any transient despite diagnostic problems which may arise as in the case of the complex transient. This fundamental or functional approach indicates the safety state of the power plant. Furthermore, it can define actions required to establish a safe state without initially knowing what caused the transient. This safe state can only be achieved if the operator can monitor a selected set of parameters which represent the state of these functions.

In conclusion, the centralization of a selected set of parameters which represent the state of the safety functions presented in a trended format would aid operations personnel during nuclear power plant transients. The implementation of this concept would result in the development of a safety panel and could meet the intent of the Safety Console concept.

5.2 Control Room Display System

Group B discussed many of these issues, developed a statement of the function of the Safety Console, and raised other issues to be considered in the design of such a console.

5.2.1 Function of a Safety Console

Group B defined the function of the plant Safety Console as being to: (1) provide real time indications of plant safety status using a minimum set of fundamental parameters and displaying these in one place; (2) assist operations and support staff in detecting and diagnosing on-going events or accidents; and (3)

determine plant safety status and provide feedback to plant staff on the results of their actions.

5.2.2 Background Review

The following background materials were reviewed by Group B:

1. Regulatory Guide 1.97 and Regulatory Guide 1.47
2. NRC Guide on Operating Plant Control Room Review
3. NUREG's 0578, 0585, and 0660 (Draft 3)
4. ANS 4.5

5.2.3 General Issues to be Addressed in Designing a Safety Console

1. What are the objectives of the Safety Console?
 - (a) Simplify operator evaluation of plant status?
 - (b) Provide systems overview of whole plant?
2. For whom is the console being designed--operators or systems engineers?
3. Where is it to be located?
4. Should console implementation be evaluated as a systems problem or as a display problem?
5. Two levels of design criteria may be required for backfit and future plants.
6. What are the effects of losing the Safety Console on plant operations?
7. Is there a potential problem of operator over-reliance?

5.2.4 Design Criteria Issues

1. It is assumed that no additional transmitting devices or sensors are required by the Safety Console that are not already mandated by other requirements.
2. What is the depth of safety significance of a Safety Console?
 - (a) Should it be immune to a single failure and/or demonstrate high reliability?

- (b) What should the depth of redundancy and independence be?
- (c) To what extent does it have to be seismically qualified?
- (d) What are the environmental qualification requirements in general?
- 3. What is the impact of the console on existing instrumentation?
- 4. Should the Safety Console have the ability to do data trending on demand?
- 5. Should indications be included relating to process limits being approached or exceeded?
- 6. What should be the implementation priorities and phasing?
- 7. What are the advantages and disadvantages of digital versus analog-based systems?
- 8. Should the plant computer be used?
- 9. Should the operator be able to call-up supporting data on the console?
- 10. What kind of alarm capability should the Safety Console have?
- 11. What kind of power supply should it have?
- 12. Should any data not presently available in the control room be included on the Safety Console?
- 13. What is the fundamental parameter set required for the overview of plant safety?

5.2.5 Group B Conclusions on the Safety Console

5.2.5.1 Purpose of the Safety Console

A Safety Console would monitor a few critical parameters in a concentrated way and give a systematic view of the status of the whole plant. It is viewed as being a display system that would compliment but not replace existing displays. It would not have predictive capability, but should have trending capability. It should assist operations personnel in correcting abnormal conditions and provide feedback to corrective actions. It would be highly reliable, but not designed for immunity to a single failure.* It would most probably be indepen-

*This recommendation is different from provisions in NUREG 0585, Section 7.2 unless credit is given for existing control room instrumentation not part of the Safety Console.

dent from the process computer. It would key the operator to look for trends and system responses using other instrumentation currently available in the control room.

5.2.5.2 Operator Training to Avoid Over Reliance.

Over reliance on the Safety Console should be avoided by means of operator training. The Safety Console should be designed to compliment, not replace, the existing control room instrumentation. Careful consideration should be given to the display location and format to optimize its accident detection and assessment function and provide minimum distraction. Plant operators should actively participate in the preparation of procedures for the operational use of this display. Such procedures should provide for sufficient use of the display during normal operation to develop operator familiarity and confidence. However, the operator should continue to place primary reliance for routine operations on the existing control room instrumentation designed for that purpose.

Operator training should emphasize the fact that the Safety Console does not provide new or more reliable information but has located and reformed existing information (parameters) for the specific purpose of early detection of accidents and assessment of plant safety.

5.2.5.3 Evaluation and Use of the Safety Console

The Safety Console concept should not be viewed as the "be-all and end-all" to plant safety. It should not run the plant and it should not be designed as a miniature control panel. The Safety Console would necessarily be incomplete and could not cover all situations.

5.2.5.4 Who is the Console for?

The console should be made available to operators as well as supervisors. Two or three minds analyzing the information presented are better than one.

5.2.5.5 State Vector

The use of the term "state vector" should be dropped in favor of a more common term. The mystery implied by the term state vector is not warranted.

5.2.5.6 Parameter Set

Industry (NSAC, AIF, etc.) should set up a dialogue with NRC to establish a fundamental parameter set for the Safety Console.

5.3 General Session Discussion on the Safety Console

Most agreed that the console's major purpose is detection and that the operator must go to other instruments for the cause. After an abnormal or accident situation has been detected, it was felt that the Safety Console should only be used to provide a broad overview of the situation. Normal control instrumentation should be used for actions to be taken. Thus, the interface between the console and other instruments is critical and requires careful human factors engineering.

The role, function, and need for a Safety Console did not achieve unanimity. Some thought if the console is good enough for detection, it should be good enough for operation. They thought that the console should go beyond detection. Others felt it was not clear that a console is needed at all and that the first question should be: "Does the console help the operator?" If it does not, it will do him no more good than current indications on the control board. Many, however, believed that by establishing a minimum set of parameters covering essential safety functions, a safety console could be developed that would assist the operations staff by: (1) providing a real time indication of plant safety status and, (2) providing feedback on the results of actions taken.

6. SAFETY SYSTEM STATUS MONITORING

NUREG 0660, draft dated March 5, 1980 states that:

NRR will require all licensees and applicants not presently committed to the requirements of Regulatory Guide 1.47 "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," to assess the need to monitor and verify operations, test, and maintenance activities by means of an automatic status monitoring system such as that described in Regulatory Guide 1.47.

Group B addressed the issues raised by NUREG 0660 in reference to Regulatory Guide 1.47. The group believed that taking the language of NUREG 0660 at face value, there is no indication that more automation will be required in implementing Regulatory Guide 1.47. NUREG 0660, draft dated March 5, 1980, states that plants not previously committed to Regulatory Guide 1.47 will be required to assess the need for a Regulatory Guide 1.47 type system. One could conclude by implication that plants presently committed will not be effected, which may be wishful thinking. The schedule as now stated in NUREG 0660 is that all licensees are to complete their assessments by September 1981, NRC is to complete its review by January 1982, and final implementation is scheduled to be completed by September 1982.

The group discussed a number of issues concerning safety system status monitoring. The concerns raised were not with the intent of the guide, which was generally supported, but with the extent -- how much automation is needed and required?

6.1 Background Review

The following background material was reviewed by Group B:

1. NUREG 0660 (Draft 3)
2. Regulatory Guide 1.47

6.2 Group B Discussion of Safety System Status Monitoring

6.2.1 Problems with Automatic Status Monitoring

A number of problems were identified when discussing the installation of an automatic status monitoring system. There is a misconception about the technical ability to undertake such a task though the technology exists. The fundamental question does not appear to be technological limitations; rather, it is the extent of technological application that can or should be required. There are many valves that would not be instrumented if such an automated system were installed. The cost of putting in a large number of additional instrumentation lines to achieve a significant level of completeness seems to be an overriding factor in this regard.

6.2.2 Manual Systems

There are ways other than automation to solve the potential human error problems of valve line-up. A system in which there is manual or semi-automatic interface with the computer to enter valve status monitoring can be instituted such as is currently being implemented by Philadelphia Electric. A code can be imprinted on all the valves and read by a hand-held automatic penlight device periodically. This is better than a clip board and should result in fewer human errors. Another way to see if a system is lined up, rather than go through individual valve indications on many valves, is to monitor key valves only.

6.3 Group B Conclusions Regarding Safety System Status Monitoring

6.3.1 Maintenance

The interface between the maintenance center and operating center of a power plant is a major plant problem. The entire question of automatic safety system status monitoring with reference to this problem needs thorough and careful analysis.

6.3.2 Plant Availability

In addition to safety considerations, it was recommended that safety system status monitoring offers significant advantages in eliminating inadvertent technical specification violations. Such a system could offer improvements in plant availability.

7. TECHNICAL SUPPORT CENTER (TSC)

NUREG 0660, draft dated March 5, 1980, states that

"In the near term, licensees and applicants will establish a Technical Support Center as described in NRR letters of October 30 and November 9, 1979. In the near term, the center will be established, provisions made for planning procedures, staffing and communications and a plan and schedule will be submitted for final upgrading of the center to specifications given in the NRR letters.

Groups B and C addressed issues concerning the Technical Support Center which NRC is asking be implemented in operating plants by January 1, 1981. The specific NRC requirements are not clear and the questions raised about the TSC relate to those discussed by the other groups:

1. What is the function of the TSC and who is to use it?
2. What parameters should be monitored in the TSC?
3. How should the TSC be integrated with existing systems?
4. What kind of equipment and systems are required?

The two groups differed somewhat in their conclusions regarding data acquisition and it was generally agreed that the issues raised need more extensive consideration if the TSC is going to serve a useful function in operating plants.

7.1 Definition of the Technical Support Center

Groups B and C used the following as their definition of the TSC: "A dedicated 'Technical Support Center' will provide a place, in close communication with the control room so as to have sufficient knowledge of current and projected plant status, for management and technical personnel to support reactor control functions, to evaluate and diagnose plant conditions, and for a more orderly

conduct of emergency operations." (NUREG 0660, Task III, p. A-1.4).

7.2 Background Review

Both Group B and Group C reviewed the following background material:

1. NUREG 0578, 0585, 0660
2. Denton letter of October 30, 1979
3. ANS 4.5

7.3 Group B Conclusions on the Technical Support Center

Group B developed a scope for the Technical Support Center that included (1) data acquisition, (2) display objectives, (3) the technical direction and assistance required to achieve these objectives, (4) the type of equipment required, and (5) uses for training.

7.3.1 Data Acquisition

Through either a dedicated analog or a multiplexed system, the TSC should have the capability to receive all plant parameters, meteorological information, and information on plant releases.

7.3.2 Data Display

Data should be displayed in the TSC to provide the following: (1) groups of plant parameters; (2) trend capability; (3) assistance in the assessment of the type of accident and suggested diagnostic procedures; (4) post-mortem information for recovery.

7.3.3 Technical Direction and Assistance

In order to provide technical direction and assistance to TSC personnel the following should be available in the TSC: (1) data base for possible analyses of thermodynamic and nuclear effects; (2) access to historical data for prediction of instrument behavior when subjected to abnormal ambient conditions; (3) "as is" mechanical and electrical drawings; (4) vendor prints and instruction

manuals; (5) directory of all safety related equipment vendor contacts; (6) procurement records of materials and components; (7) plant operating procedures; (8) emergency plans; (9) TSC equipment operating procedures; (10) data link protocol.

7.3.4 Equipment

The various functional requirements for display, recording, processing, and transmission of plant data should be performed by a single, integrated, non-Class 1E computer-based system. Dedicated software and I/O devices should provide the specific functions described in 7.3.1 and 7.3.2. The system should have proven reliability/availability, comparable to that of the plant Class 1E Systems. It should not, however, be subject to the single failure criteria as defined in IEEE 279. Inputs to the system from Class 1E circuits should be isolated in accordance with the existing plant designs. If a Safety Console is developed, an identical data display system should be made available in the TSC.

7.3.5 Use of TSC System for Training

The TSC can be used to improve operator-machine performance using simulated accident sequences for training. These exercises would insure effective and timely use of the display system during emergency operations and also provide realistic experience for ongoing format upgrading.

7.4 Group C Conclusions on the Technical Support Center

Group C limited their scope to discussing the Technical Support Center in terms of data acquisition only. All other physical aspects of the TSC were excluded. Long-range plans were considered without reference to any specific time schedule for implementation which may be imposed by the NRC.

7.4.1 Parameter Selection

It was recommended that ANS 4.5 "Accident Monitoring Capability for the Control Room Operator in a Nuclear Power Generating Station," be considered in the

selection of parameters needed by the Technical Support Center for technical management focus.

7.4.2 Guideline for Parameter Selection

A guideline that should be followed is that only data which would be supportive of the Technical Support Center, as defined in NUREG 0660, III-A-1.4, should be included. Only technical management data needs to be included. Variables from ANS 4.5 which are exclusively Type A should be excluded from the Technical Support Center data list. Types B, C, and D variables from ANS 4.5 plus some off-site radiological and meteorological parameters should be included. (See Appendix B, ANS 4.5, for definition of the variable types.)

The data set derived by following the above guideline is not intended to be all-inclusive, but rather is intended to support the most likely short-term activities that will be conducted from the Technical Support Center. Techniques for securing more detailed data (without interference to control room activities) for long-term analysis should also be studied, e.g., telephone communications and runners.

7.4.3 Administrative Control

Clearly regulate control room/TSC communication with the operator being responsible. The communication should not interfere with control room activities.

7.5 General Session Discussion on the Technical Support Center

The Technical Support Center is still an open question. The TSC requirements are vaguely defined by NRC; therefore, the industry has a good opportunity to present its recommendations for consideration. Some believed that the minimum parameter set developed by Group D (see Section 4) could serve as a basic set of data needed in the TSC, although the need for additional data for performing diagnostic evaluations was recognized.

8. NUCLEAR DATA LINK

NUREG 0660, draft dated January 23, 1980, states that:

"Nuclear data link" is the term given to a system that will remotely access facility data and transmit the data and display information in the NRC Operations Center. The information will allow NRC to analyze and evaluate the plant situation in emergency conditions and to develop or evaluate proposed accident-mitigating actions.

The Nuclear Data Link systems design evaluation is being conducted for NRC by Sandia Laboratories and has progressed to the point where preliminary cost estimates are being undertaken. No implementation schedule is yet available. (See letters from NRC in Appendix G.)

Group C discussed the Nuclear Data Link and raised a number of issues.

8.1 Background Review

Group C reviewed the following material related to the Nuclear Data Link:

1. NUREG 0660, (Draft 2)
2. NRC letter from Bundnitz dated March 6, 1980 (Appendix G)
3. NRC letter from Eisenhut dated March 12, 1980 (Appendix G)

8.2 Scope Statement

Group C was not aware of a specific purpose for the data link. Since NRC has not yet defined their intended use of the data, they were unable to compile a specific data list.

8.3 Issues Raised by Group C on the Nuclear Data Link

1. There was no consensus on the technical basis for the data link; however, if a link is imposed, the data requirements should not be any greater than a subset of that required for the Technical Support Center.

2. Some distinction should be made regarding the real time data requirements during normal conditions vs. requirements following accident initiation.
3. Consideration should be given to the security of proprietary data.
4. The implementation schedule for a data link needs to be coordinated with other related data acquisition requirements (e.g., Safety Console, Technical Support Center, Emergency Operations Center).
5. Data availability and data conversion is a critical issue and should be further pursued through the questionnaire mentioned in Eisenhower's letter of March 12. (Appendix G)

8.4 General Session Discussion on the Nuclear Data Link

There were a number of concerns raised regarding the Nuclear Data Link. First, the scope of the proposed system is a problem. No specific purpose for the link was defined in Eisenhower's letter of March 12, 1980, nor does his letter give any timetable. Another concern raised was about the security and use of the data. Who has accessibility to the data? NRC is asking for real time data. Is the fundamental machinery of government set up for this? Is there an administrative, legal, and technical structure to handle it? Are there legal questions relating to these issues that need to be addressed? There was concern about equipment required, specifically the sensors; most plants do not have the instrumentation in place to meet projected NRC requirements for a Nuclear Data Link. A last concern was in regards to the data format. Who should assume responsibility for conversion of data to a fixed format if required? If plants have to convert data to a fixed format it could be expensive.

8.5 General Session Conclusions on the Nuclear Data Link

1. Industry should undertake an indepth review of the NDL -- the data flow, formatting, access, and how protocols are to be handled.
2. Industry should work with NRC to define requirements for a NDL that is technically and economically feasible and meets safety needs.

9. GENERAL RECOMMENDATIONS

Recommendation 1. The nuclear industry should take the initiative in defining functional requirements and the best means for implementing data acquisition and display systems. A concerted, coordinated industry review and response to safety issues and NRC requirements could stop the piecemeal, plant by plant, nonstandardized approaches to implementing safety guidelines.

Recommendation 2. There should be a continued dialogue among utilities, vendors, and other industry organizations to reach consensus positions with respect to data acquisition and display systems. The NRC safety improvements program appears fragmented at this time at the local utility level and the priorities in the safety area are not always clear. In practice, the NRC sets policy in broad terms and the details are hammered out in the review process with each plant. Details come out in a sifting process. It is believed that NRC would benefit from industry input.

Recommendation 3. The NSAC workshop participants conducted an indepth review of NRC TMI-related requirements and had difficulty in understanding the functional and technical requirements of some of the guidelines. It is recommended that industry should charge a group like NSAC with the responsibility to establish a dialogue with NRC to discuss and identify the technical rationale underlying some of the new requirements. It is desirable to develop a publicly available technical rationale for new requirements before implementation proceeds. Otherwise, there is no way to adequately critique what is being proposed.

A dialogue would be particularly useful in regards to the Safety Console. NRC has not yet developed functional or technical specifications for the Safety

Console and it is important that the nuclear industry have input before specifications are frozen. NRC and the industry need agreement on critical issues if a functionally and technologically superior system is to be developed.

Recommendation 4. NSAC should establish a list of functions and associated fundamental parameter sets for the Safety Console, the Technical Support Center, and the Nuclear Data Link and other proposed systems, facilities, and communication flows. In addition, it should establish a coordinated approach to Safety System Status Monitoring (Regulatory Guide 1.47) and Regulatory Guide 1.97.

Recommendation 5. The industry should review the implementation schedule for all TMI related requirements with the NRC. The schedule for implementation of many TMI related requirements is unrealistic.

REFERENCES

1. NUREG 0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants Reported to Congress January 1979."
2. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979, as clarified and discussed by letter from H. R. Denton to All Operating Nuclear Power Plants, October 30, 1979.

Section

2.1.3.b - "Instrumentation for Detection of Inadequate Core Cooling"

3.1.8.b - "Increased Range of Radiation Monitors"

ACRS Instrumentation - "Containment Pressure Indication"

"Containment Water Level Indication"

"Containment Hydrogen Indication"

2.2.2.b - "Onsite Technical Support Center (TSC)" both in instrumentation and data link communications.

3. NUREG-0585 - "TMI-2 Lessons Learned Task Force Final Report, Appendix A - Final Recommendations of TMI-2 Lessons Learned Task Force," October, 1979.

Section

7 - "Man-Machine Interface"

7.1 - "Control Room Reviews" (items 1 through 9)

7.2 - "Plant Safety Status Display" (Safety State Vector)

7.3 - "Disturbance Analysis Systems" (Operator Warning and Guidance System)

4. NUREG-0660 (Draft - 3/05/80), "Action Plans for Implementing Recommendations of the President's Commission and Other Studies of TMI-2 Accident."
5. NUREG-0610, "Draft Emergency Action Level Guidelines for Nuclear Power Plant", September 1979.
6. "Report of the President's Commission on the Accident at Three Mile Island," October 30, 1979.

Commission Recommendations

A.4.c, C.3.d, and D.1 based on Commission Findings A.8.a, A.8.b, and G.8.e. related to plant data recording, display, and analysis and control room design.

7. "Preliminary Data Requirements for Nuclear Data Link (NDL)", H. R. Denton and V. Stello, Jr., to S. Levine - Director, Office of Nuclear Regulatory Research, November 21, 1979, letter from R. H. Budnetz, March 6, 1980, and letter from D. E. Eisenhut, March 12, 1980.
8. Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems", May, 1973.
9. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," proposed Revision 2, December, 1979.
10. American National Standard (ANS) 4.5, "Accident Monitoring Capability for the Control Room Operator in a Nuclear Power Generating Station, November 1979."

APPENDIX A

Letter from the AIF Policy Committee
on follow-up to the Three Mile Island
Accident, dated March 14, 1980

ATOMIC INDUSTRIAL FORUM

Subject Objective of NSAC Workshop on Nuclear Plant Safety Parameters

Date March 14, 1980

To Byron Lee, Jr., Chairman, AIF Policy Committee on Follow-Up
to the Three Mile Island Accident
From Stephen Howell, Chairman, Subcommittee on Control Room
Considerations
Warren Owen, Chairman, Subcommittee on Emergency Response
Planning

At the March 11, 1980 meeting of the Subcommittee on Control Room Considerations, the objective of the upcoming NSAC Workshop was reviewed. Subsequently the two of us and Bob Szalay discussed this in the context of the overall industry safety efforts.

In your absence, we concurred with the Workshop objective, which is to establish a common minimum set of parameters for nuclear power plant safety assurance and supervisory overview.

These parameters might be integrated into a safety panel design or utilized in some other control room evolution developed through design reviews now underway. This minimum set could be instrumental in avoiding escalations in requirements that would result from the absence of this coordinated approach.

Both of our subcommittees can incorporate the workshop results into their respective reviews of numerous pending issues which have some commonality in functional objectives. For example, related common functional objectives exist among the issues of control room design reviews, habitability, plant safety parameter display console, instruments for monitoring accident conditions (R.G. 1.97), and safety system monitoring. Similarly, the onsite technical support center, off-site centers, emergency planning criteria, and the data-link issues have elements in common with the safety parameter exercise.

Following the Workshop and preliminary subcommittee discussions, we should be able to evaluate the usefulness of an integrated approach to maximizing safety benefits while limiting regulatory requirements that duplicate the same functional objective and waste scarce resources.

We would like to make this a principal agenda item at the upcoming March 27, 1980 Policy Committee meeting.

cc: Ed Zebroski, Director,
Nuclear Safety Analysis Center

INTEROFFICE MEMORANDUM INTEROFFICE MEMORANDUM INTEROFFICE MEMORANDUM

APPENDIX B

Summary of the NRC TMI Regulations

NUREG 0660
NUREG 0585
NUREG 0578
NUREG 0510
Regulatory Guide 1.97
Regulatory Guide 1.47
ANS 4.5

NUREG 0660
Action Plans for Implementing Recommendations
of the Presidents Commission and Other
Studies of TMI-2 Accidents
Draft March 5, 1980

Task 1.D Control Room Design

<u>Action</u>	<u>Status/Schedule</u>
1. <u>Control Room Design Reviews</u> NRR will require that operating reactor licenses and applicants perform a detailed control room design review to identify and correct design deficiencies.	Essex Corporation reviewing guidelines to be issued May 1980. Plant reviews to be completed by 9/81.
2. <u>Plant Safety Parameter Display Console</u> NRR will require all licenses and applicants to install a safety parameter display system that will display to the operator a minimum set of parameters (safety state vector) which define the safety status of the plant.	NRR requirements will be issued 6/80. Plants to submit plans to NRR by 1/81.
3. <u>Safety System Status Monitoring</u> NRR will require all licenses and applicants, not presently committed to Reg. Guide 1.47, to assess the need to monitor and verify operations, test, and maintenance activities by means of an automatic status monitoring system such as described in Reg. Guide 1.47.	Plants to complete assessments by 9/81. NRR will complete reviews of license assessments by January 1982.
4. <u>Control Room Design Standard</u> SD will issue for comment a proposed regulatory guide based on an evaluation of industry standards (IEEE 566 and 567)	Reg. Guide to be issued by SD by July 1981 for comment. SD will develop implementation schedule and Reg. Guide by May 1982. NRR will ensure compliance by May 1983.
5. <u>Improved Control Room Instrumentation Research</u>	
(a) Operator-process communication	Report 12/80
(b) Plant status monitoring	Report 12/80
(c) On-line reactor surveillance system (ORNL)	Field test 10/81 Report 7/80
(d) Process monitoring instrumentation	Design Criteria 8/80
(e) Disturbance analysis system (LOFT)	Verified by 12/82

NUREG 0660
Task III.A.1
Improve Licensee Emergency Preparedness--Short Term

<u>Action</u>	<u>Status/Schedule</u>
1. <u>Upgrade Emergency Preparedness</u> Six NRC teams are reviewing licensee state, and local capabilities and plans operating reactors are to upgrade their emergency plans.	Review and upgrade to be completed by 8/80. By 9/81 an integrated assessment of the implementation of plans will be completed.
2. <u>Upgrade Licensee Emergency Support Facilities</u>	Letters issued 9/13/79 and 10/30/79 requiring TSC. Near term requirements were to be completed by 1/1/80. Final action to upgrade the center will be completed by 1/1/81.
(a) <u>Technical Support Center (TSC)</u> A TSC is to be established at each plant to provide the main communication link between the plant, the Operational Support Center, and the NRC in emergencies.	
(b) <u>Operations Support Center (OSC)</u> The OSC is to be established separate from the control room as a place in which operations support personnel assemble and report in an emergency situation to await instructions from the operating staff. The OSC is to be provided with communications with the plant control room, Technical Support Center, and Emergency Operations Facility.	OSC to be established by 1/1/80 as described in 10/30/79 letter from NRC.
(c) <u>Emergency Operations Center (EOC)</u> An EOC is to be established and sized and equipped to function as (1) a center for the licensee's command and control functions of onsite operations and evaluation and coordination of all licensee activities, onsite and offsite, related to an emergency having actual or potential environmental consequences; and (2) a near-site central command post for appropriate representatives of Federal, State, and local governments. Included in the functions of the facility will be the provision of information regarding current and projected plant status. Some press facilities will also be provided.	The requirement for an EOC exists in Reg. Guide 1.101. Near terms requirements were to be completed by operating plants 1/1/80. Final action to upgrade the facility is to be completed by 1/1/81.

NUREG 0660
Task III.A.2
Improving Licensee Emergency Preparedness--Long Term

<u>Action</u>	<u>Status/Schedule</u>
1. Amend 10CRF50 and 10CFR50 Appendix E.	6/30/80 Commission paper to be completed. Plants to implement rules 6 months after issued.
2. Development of Guidance and Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in support of Nuclear Power Plants.	NRC/FEMA criteria published in Federal Register. A NUREG Report will be published 1/30/81 providing guidance on test exercise scenarios. Plants will participate in development of criteria.

Task III.A.3
Improving NRC Emergency Preparedness

<u>Action</u>	<u>Status/Schedule</u>
1. Develop NRC role in responding to nuclear emergencies.	MITRE Corporation is undertaking a study to be completed by March 1980. A commission paper will be completed by May 1, 1980. Implementing procedures to be incorporated into IE manual by June 1980.
2. Improve Operations Centers (OC) The NRC Operations Center in Bethesda will be upgraded to support activities in response to a major accident.	Physical requirements and arrangement for OC to be completed 3/80. Regional OC modifications to be completed January 1981.
3. <u>Communications</u>	
Direct dedicated telephone lines (OPX) to be installed at each operating power plant to NRC.	To be installed by 2/80.
A study of National Warning System (NAWAS) to be undertaken.	No date given.
High Frequency (HF) radios will be installed at each site to ensure uninterrupted communication during events such as earthquakes, hurricanes, floods, and sabotage.	Study on back-up radio communication to be completed by 7/1/80.
NRR will obtain authorization for teletype circuits with the National Oceanic and Atmospheric Administration (NOAA).	

NUREG 0660
Task III.A.3 (continued)
Improving NRC Emergency Preparedness

<u>Action</u>	<u>Status/Schedule</u>
4. <u>Nuclear Data Link (NDL)</u> A system is to be developed that remotely accesses facility data and transmits the data and displayed information in the NRC Operations Center.	Sandia is developing the concept and upgrading the NRC operations Center. NDL Phase I Study to be completed 4/1/80. Implementation schedule to be defined.
5. <u>Training, Drills and Tests</u> Headquarters drills and exercises presently being conducted will continue and a schedule for frequency developed.	Continuing schedule to be completed in connection with routine budgetary processes.
6. <u>Interaction of NRC with Other Agencies</u>	
(a) <u>International</u> Agreements with Canada and Mexico for mutual cooperation and assistance during emergency events.	12/80
(b) <u>Federal</u> Overall Federal response plans to be developed.	FEMA determined schedule.
(c) <u>State and Local</u> NRC will inform state and local agencies of its role and all emergency procedures to be followed.	4/80

Identification of Unresolved Safety Issues Relating to Nuclear Power Plants
Reported to Congress January 1979

ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT

<u>Action</u>	<u>Status/Schedule</u>
<p>In order to assure that electrical equipment in safety systems will perform its function under accident conditions, the NRC requires that such equipment--principally equipment associated with the emergency core cooling system and containment isolation and cleanup systems--be "environmentally qualified." Specific electrical equipment of concern during postulated accident sequences includes:</p> <ol style="list-style-type: none"> 1. The instrumentation needed to initiate the safety systems and provide diagnostic information to the plant operators (e.g., electrical penetrations into containment, any electrical connectors to cooling which transmits signals, and the instruments themselves. 2. Control power to monitor operators for certain valves (e.g., ECCS and containment isolation valves located inside containment. 3. Fan cooler motor for those plants that utilize fan coolers for containment heat removal. 	<p>Generic Task A-34 Instrumentation for Monitoring radiation and process variables during accidents.</p> <p>NRC is developing criteria and guidelines to be used by applicants, licenses, and staff reviewers to support implementation of Key Guide 1.97, Revision 1.</p>

TMI-2 Lessons Learned Task Force
Status Report and Short-Term Recommendations
July 1979
Instrumentation for Deletion of Inadequate
Core Cooling in PWRs and BWRs (Section 2.1.3.6)

1. Licenses shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation.

Each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition.

2. Licenses shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices...giving an unambiguous, easy-to-interpret indication of inadequate core cooling.

"The purpose of this recommendation is to provide the reactor operator with instrumentation, procedures, and training necessary to readily recognize and implement actions to correct or avoid conditions of inadequate core cooling."

NUREG 0585
TMI-2 Lessons Learned Task Force Final Report
October 1979

Action

Status/Schedule

7.2 Plant Safety Status Display

Each licensee should be required to define and adequately display in the control room a minimum set of plant parameters (in control terminology, a state vector) that defines the safety status of the nuclear power plant. The minimum set of plant parameters should be annotated for sensor limits, process limits, and sensor status. The annotated set of plant parameters should also be available in real time by a reliable, single-failure-proof system located in the control room. The annotated set of plant parameters should also be available in real time in the onsite Technical Support Center (TSC).	Implementation by January 1, 1981, in consonance with the final implementation date for the onsite Technical Support Center recommended in NUREG-0578.
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As a further guideline for the development of the safety state vector, the status of the plant process should be designed and instrumented as a function of the various barriers against release of radioactivity. For example, the two primary barriers are the fuel cladding and the reactor coolant pressure boundary. Thus parameters such as primary liquid inventory, and coolant radioactivity levels would be principal components of the state vector for these levels of defense. Similarly, reactor coolant level, containment water level, containment hydrogen content, etc., would be principal components of the state vector for the engineered safety feature levels of defense.

Regulatory Guide 1.47
Bypassed and Inoperable Status Indication
for Nuclear Power Plant Safety Systems
May 1973

<u>Action</u>	<u>Status/Schedule</u>
1. Administrative procedures should be supplemented by a system that automatically indicates at the system level the bypass or deliberately induced inoperability of the production system and the systems actuated or controlled by the protection system.	In NUREG 0660 draft, it states that NRR will require all licenses and applicants not presently committed to Reg. Guide 1.47 to assess the need to monitor and verify operations, test and maintenance activities by means of an automatic status monitoring system. Plant assessments to be completed by 9/81. NRR will complete reviews of license assessments by 1/82.

Regulatory Guide 1.97

Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

Proposed Revision 2 of December 1979

<u>Action</u>	<u>Status/Schedule</u>
<u>Criterion 13</u> , "Instrumentation and Control of Appendix A," General Design Criteria for Nuclear Power Plants, in 10CFR50, requires that instrumentation be provided to monitor variables and systems for accident conditions as appropriate to ensure adequate safety.	REG. Guide 1.97 is scheduled for final issue 8/80. The NRC will send a letter to each utility indicating specific requirements for each plant after August 1980. Full implementation of Reg. Guide 1.97 is presently scheduled for June 1982.
<u>Criterion 19</u> , Control Room, in 10CFR50, requires that a control room be provided from which actions can be taken to maintain the nuclear plant in a safe condition under accident conditions, including loss-of-coolant accidents, and that equipment, including the necessary instrumentation, and appropriate locations outside the control room be provided with a design capability for prompt hot shutdown of the reactor.	
<u>Criterion 64</u> , Monitoring Radioactive Releases, in 10CFR50, requires that means be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluid, effluent discharge packs, and the plant environs for radioactivity that may be released from postulated accidents.	
Guide 1.97 describes a method acceptable to NRC for complying with these regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant.	
<u>Variables/Parameters</u> , Reg. Guide 1.97 provides a list of variables (Tables 1, 2, and 3) which should be used in developing the minimum set of instruments and their respective ranges for accident-monitoring systems. Table 1 lists design criteria, Table 2 lists PWR variables and Table 3 BWR variables--their ranges, type, and purpose.	

American National Standard 4.5
Accident Monitoring Capability for
the Control Room Operator in a
Nuclear Power Generating Station

November 1979

ANS 4.5 delineates criteria for determining the variables to be monitored by the control room operator of a light water reactor, as required for safety, during the course of an accident and during the long-term stable shutdown phase following an accident. The Standard was prepared with two primary objectives: (1) to address that instrumentation that permits the operator to monitor expected parameter changes in an accident period and (2) to address extended range instrumentation deemed appropriate for the possibility of encountering previously unforeseen events.

The Standard defines five classifications of variable types for the purpose of aiding the designer in his selection of accident-monitoring instrumentation and applicable criteria:

- Type A: Variables that provide information needed for preplanned operator actions. (Plant-specific) variables that provide information to indicate whether plant safety functions are being accomplished.
- Type B: i.e., reactivity control, core cooling maintaining reactor coolant system, integrity, and containment integrity, and radioactive effluent control.
- Type C: Variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product release, i.e., fuel cladding primary coolant pressure boundary, and containment.
- Type D: Variables that provide information to indicate the performance of individual safety systems.
- Type E: Variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and for continuously assessing such releases for providing defense in depth, and for diagnosis.

Time phases (Phases I, II, and III) delineated in ANS-4.5 are not specified for each variable because they are plant specific.

The scope of the standard is limited to onsite environment and process monitoring.

APPENDIX C

NSAC Workshop Attendees
Working Group Participants
Agenda

LIST OF ATTENDEES

NSAC DATA ACQUISITION AND DISPLAY SYSTEMS WORKSHOP
DALLAS, TEXAS

MARCH 18-19, 1980

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WORKSHOP ON
DATA ACQUISITION AND DISPLAY SYSTEMS
WORKING GROUP TOPICS

MEMBERS

GROUP A - EXISTING SYSTEMS UPGRADE

Control Room Design Review
Shift Technical Advisor Needs
Regulatory Guide 1.97
Alternative System Architectures
(including Process Computer)

* Jim Easter, Westinghouse

Ellery Hammond, Iowa Electric
Jim Osborne, FP&L
Chuck Johnson, PSE&G
Dick Kanazawa, EPRI
Ripley Newcomb, VEPCo
Wayne Alphonso, LP&L

GROUP B - CONTROL ROOM DISPLAY SYSTEM

Safety Status Panel
Safety System Status Monitoring
Regulatory Guide 1.47

* Tony Buhl, TEC

Bob Hamilton/D. Fischer, GE
Roger Newton/G. Daniels,
Wisconsin Electric/RG&E
E. S. Patterson, B&W
Lester Williams, S&W
Dick Squires, Commonwealth Edison
Melvin Silliman, Niagara Mohawk

GROUP C - REMOTE DATA TRANSMISSION

On-Site Technical Support Center
Emergency Control Center
NRC Nuclear Data Link
Engineering Data to Vendors
"Reactimeter"/Data Logger

* Ted McMeekin, Duke Power

T. G. Shultz, C-E
Tag Rogers, PG&E
Larry Kidd, VEPCO
Don Scapini, Quadrex
Srivatsan Nadipuram, Con. Edison
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GROUP D - PARAMETER SET INTEGRATION

* Bill Gordon, Bechtel

Les Oakes, ORNL/NSAC
James C. Robinson, TEC
Mike Burzynski, TVA
Russ Klatt, PG&E
Ralph Phelps, So. Cal. Edison
Allan S. Topor, Toledo Edison

* Working Group Leader

AGENDA

WORKSHOP ON

DATA ACQUISITION AND DISPLAY SYSTEMS

-- Directions after TMI - 2

March 18-19, 1980

TUESDAY, MARCH 18, 1980

8:30 - 9:15	Opening Remarks
9:15 - 10:15	Review of Regulatory Requirements (A. Buhl, TEC)
10:15 - 10:30	DASS Technical Advisory Group Meeting (D. Wilkinson, NSAC)
10:30 - 10:45	Break
10:45 - 11:15	Westinghouse Owner's Group Response - AD HOC Committee (R. Newton, Wisconsin Electric Power)
11:15 - 11:45	General Electric Owner's Group Control Room Subcommittee Report (Ellery Hammond, Iowa Electric Light & Power)
11:45 - 12:15	Report on Near Term Operating License (NTOL) Plants
12:15 - 1:00	Lunch (Catered)
1:00 - 1:30	Data Acquisition and Display Improvements -- The Plant Operator's Viewpoint (G. Caldwell, Duke Power)
1:30 - 2:15	General Discussion and Work Group Directions
2:15 - 5:30	Work Group Session (I)
5:30 - 6:00	Work Group Status Reports
6:30 - 7:30	Dinner (Catered)
8:00 - 9:30	Work Group Session (II)

WEDNESDAY, MARCH 19, 1980

8:00 - 8:30	Summary Remarks
8:30 - 10:00	Work Group Session (III)
10:00 - 10:15	Break
10:15 - 11:15	Work Group Reports
11:15 - 12:00	General Discussion
12:00 - 1:00	Lunch (Catered)
1:00 - 3:00	General Session - Integrating It All Together
3:00	Adjourn

APPENDIX D

Schemata Presented By

Westinghouse Owner's Group Ad Hoc Instrument
Systems Committee for a Technical Support Center

FIGURE 1 FUNCTIONAL BLOCK DIAGRAM
SINGLE NON-CLASS 1E SYSTEM

Rev. 2
2/14/80

APPENDIX E

General Electric Owner's Group Control Room Committee
Schematic of a Data Link System Being Installed at PG&G

FUNCTIONAL BLOCK DIAGRAM

DATA LINK SYSTEM

APPENDIX F

AIF Policy Committee on Follow-up to the Three Mile Island Accident

Schemata of an Emergency Response Organization

Figure F-1. Typical Emergency Response Organization

Figure F-2. Facilities and Communications Flow

APPENDIX G

Nuclear Data Link

NRC Letter from Denton and Stello, November 21, 1979
NRC Letter from Budnitz, March 6, 1980
NRC Letter from Eisenhut, March 12, 1980

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

NOV 2 1 1979

MEMORANDUM FOR: Saul Levine, Director, Office of Nuclear Regulatory Research

FROM: Harold R. Denton, Director, Office of Nuclear Reactor
Regulation
Victor Stello, Jr., Director, Office of Inspection
and Enforcement

SUBJECT: PRELIMINARY DATA REQUIREMENTS FOR NUCLEAR DATA LINK (NDL)

SUMMARY

The NRC Operations Center is being upgraded to provide the Commissioners and Headquarters functional groups with a centralized operations center for the purpose of crisis management and technical support in the event of a nuclear reactor accident. A preliminary information paper is being prepared by IE for discussion with the Commissioners by the end of November 1979 for early feedback on functional requirements.

This memorandum contains a draft specification for real-time data required from PWR's and BWR's by the NRC Operations Center in order to execute the regulatory function during nuclear incidents. The data are to be transmitted by means of a communications link from the control room/onsite Technical Support Center at the power plant to the NRC Operations Center.

The data specified herein is of sufficient scope to support the Phase I design goal of the Operations Center consisting of limited real-time data for display and evaluation of plant operational safety. We anticipate that some refinements, modifications and additions to the data list will be required in order to achieve the Phase II goals for the Operations Center. These alterations will be addressed at a future date.

The plan will be to provide these data requirements to the industry task group to facilitate the coordinating functions with each plant. The IE Regional Offices will participate in determination of plant-specific data requirements as necessary.

SCOPE

The purpose of this memorandum is to define the type of data needed at the Operations Center and the specifications for the data. Once defined, the data specifications are to be used for the design of a communications link between nuclear power plants and the NRC. Furthermore, the data specifications are only for BWR and PWR facilities and does not address fuel processing facilities, experimental facilities and HIER's. Sandia Laboratories has been contracted to conduct a feasibility study regarding the communication link.

CONTACT: R. W. Woodruff, IE (49-28180)
L. Beltrachi, NRR (49-27233)

GENERAL REQUIREMENTS

1. PLANT IDENTITY: Transmissions of data shall contain plant and unit identity.
2. DATE TAGS: Transmission of data shall be dated (month, day, year) to record the date when data was sampled.
3. TIME TAGS: Each parameter in the data file shall be time tagged (hours, minutes, seconds, local time) to record the time when the data was sampled.
4. EVENT DISCRIMINATOR/DATA DISCRIMINATOR: In order to obtain maximum use of the discrete data transmitted over the link, a data sample frequency must be chosen so as not to miss an important event. However, a short data sample frequency may result in a large amount of useless data. We, therefore, recommend that the transmitting end of the data link be capable of varying the data sample frequency automatically on the occurrence of specific detectable events. Provisions for manual selection of data sample frequency at the receiving end and transmittal end should also be considered in the design.
5. INITIATION OF DATA TRANSMISSION: Automatic initiation of data transmission should be studied by Sandia. The Safety Injection Signal, the Containment Isolation Signal, upon reaching the pressure setpoint for acutuation of the pressurizer safety valves (PWR's), and suppression pool temperatures greater than a specified value for BWR's should be considered as possible triggers for automatic initiation. Provisions for manual initiation of data transmission should also be considered in the design of the data link.
6. APRIORI EVENT DATA: The licensee's data acquisition system should be required to record a fixed time length of process and status data prior to activation of data transmission. These data are to be "frozen" on an event and transmitted by the data link upon initiation of data transmission. This will provide the NRC with apriori data for analysis purposes. We recommend that Sandia establish the fixed time length in their feasibility study.
7. PRIORITY OF DATA TRANSMISSION: We recommend that the Safety Vector be given top priority for transmission. This vector remains to be defined, and should be contained within the data specified herein.

Priority of data transmission is specified in terms of transmission frequency. For example, low priority data is transmitted infrequently relative to high priority data. However, it should be noted that evaluation of the data at the NRC Operations Center can only be achieved with time compatible data. Thus, real-time evaluations may have a time lag as large as the longest data-sample frequency.

8. MULTIPLE REACTOR INCIDENTS: The sizing of the communication link and the Operations Center must consider simultaneous/overlapping incidents or incidents at multiple unit sites in terms of total data requirements.
9. TWO-WAY Communication: The communication link should allow transmission of information, such as results of calculations, drawings, etc., from the NRC Operations Center to the site of the incident.

10. GROWTH FLEXIBILITY: To accommodate future refinements, upgrading of data requirements, and an increase number of reactor and fuel cycle facilities, the system should be sized to manipulate at least an order of magnitude increase in data over what is currently stated in the enclosures.
11. SCOPE OF DATA: The list of data in the enclosure was defined on the basis of information required to evaluate the status of the core cooling process from process heat source to ultimate heat sink. Furthermore, event data, status of safety systems/safety-related systems and radiological data requirements are defined for purposes of evaluating licensee's actions, status of safety systems and radiological releases. It is recognized that all plants may not be instrumented to provide the data specified. For such cases, the licensee should be requested to provide equivalent data, if available.

The listing of parameters in these data specifications does not necessarily indicate a regulatory requirement for such a measurement unless the parameter is also defined in the General Design Criteria, Regulatory Guides, NUREG 0585, and NUREG 0578. In formulating the data specifications, we consider and incorporated many of the parameters defined in the current draft Regulatory Guide 1.97, regarding accident monitoring (1.97, draft 1 to Proposed Rev. 2, dated October 15, 1979).
12. PLANT INTERFACE REQUIREMENTS: These requirements, which may contain interface requirements to safety systems per EDC 24, are beyond the scope of this report.
13. FLOW DATA, LEVEL DATA: These type of data are specified in terms of percent design flow and percent of level vs each plant has different values. Each plant must define design values for these parameters and transmit them along with the data.

DISCUSSION OF SPECIFICATIONS

The data specifications for PWR's are contained in Enclosure I. The data specifications for BWR's are contained in Enclosure II. Definitions of the pertinent headings are as follows:

NO. OF SENSORS: Specifies the minimum number of sensors for the associated parameter (note - some may be plant-dependent and vary from the number stated).

TRANSMISSION ACCURACY: Minimum resolution required of transmitted data. This item has no relationship to sensor accuracy. It is anticipated that transmission accuracy, along with range, will specify the number of bits required to transmit the data.

RANGE: This term conservatively estimates the minimum and maximum value of the desired parameter for purposes of bit specification in digitizing the data.

FREQUENCY: Sample frequency (or rate) of parameter, in terms of time. A maximum and a minimum value is defined.

Harold R. Denton
Director
Office of Nuclear
Reactor Regulation

Victor Stello, Jr.
Director
Office of Inspection
and Enforcement

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

MAR 6 1980

Mr. Warren Owen, Chairman
TMI Emergency Response Planning
Subcommittee
Atomic Industrial Forum
c/o Mr. Art Bivens
7101 Wisconsin Avenue
Washington, DC 20014

Dear Mr. Owen: *Warren*

As you know, the Nuclear Regulatory Commission (NRC) is currently engaged in developing systems design concepts for a Nuclear Data Link (NDL) to furnish data and information from licensed nuclear plants to the NRC Headquarters Operations Center in the event of incidents or accidents.

The NDL systems design effort, which is being conducted for the NRC by Sandia Laboratories, has now progressed to the point where preliminary cost estimates are being undertaken. Draft NDL specifications and preliminary NDL system concepts are being used by Sandia to scope estimates of time and funds needed to achieve operational capability of the NDL.

It is planned that implementation of the transmission and Operations Center subsystems of the NDL system would be accomplished by Sandia Laboratories with oversight by the NRC Office of Nuclear Regulatory Research and with concurrence by the Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement; while licensees would be responsible for procurement and installation of plant subsystems which would meet detailed NDL specifications.

Under these circumstances, we would like to invite AIF participation in the preparation of a cost estimate covering licensee participation in implementation of NDL. AIF engineers have participated as observers in NDL design meetings at Sandia and have been cooperative and helpful in the NDL systems definition effort. A similar participation in the cost estimation effort now underway will enhance the NRC effort to get a realistic understanding of NDL system implementation costs to the government and to the licensees, and thus provide a sound basis for decisions as to cost and pacing of system options. Further information concerning this matter is available from Sam Bassett, Project Engineer, at (301) 427-4360.

Sincerely,

A handwritten signature in black ink, appearing to read "Bob Budnitz", with a stylized flourish at the end.

Robert J. Budnitz, Director
Office of Nuclear Regulatory Research

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

March 12, 1980

Distr. to Ginna
NSARB

TO ALL OPERATING NUCLEAR PLANTS

Gentlemen:

SUBJECT: NRC NUCLEAR DATA LINK (NDL)

In response to the Kemeny Report, established to investigate the accident at the Three Mile Island Unit 2 nuclear plant, on December 7, 1979, the President issued a number of actions to improve the safety of nuclear power plants. Included in these actions was for "the NRC to evaluate the need for a federal presence in the control room of operating reactors, such as by observers or continued computer monitoring of vital plant parameters." The staff believes that the most effective and timely response to the announced initiative would be to continue in its evaluation of a computerized monitoring system of vital plant parameters.

The staff is currently engaged in improving the capabilities of its NRC Operations Center (OC) at Bethesda, Maryland, in order to provide the Commission and senior staff members with vital plant parameters and information from licensed nuclear plants in the event of incidents or accidents. Improved capabilities which are under consideration for the OC will include automatic data processing, data storage, data display and data recall capability to be achieved through the use of digital computers. The overall objective for obtaining such information is to enable the NRC, in the event of a nuclear accident at a licensed reactor facility: (1) to monitor and evaluate the situation and potential hazards; (2) to advise the licensee's operating staff as needed; and (3) in an extreme case, to be able to issue orders governing such operations.

A systems study of a digital computer based Nuclear Data Link (NDL) to bring this type of information from nuclear plants to the OC has been instituted with Sandia Laboratories. The system study has been directed toward digital communication systems because of the proven availability of the technology and the successful, reliable, operating experience, and the operating costs of well designed systems in the business community. The goal is to achieve an initial operating capability to receive, process, display, evaluate and record essential, but limited, data at the new OC installation by January 1, 1982. A subsequent program will delineate improvement and refinements to the system.

The data which are being considered for acquisition, transmission, receipt and display will consist of those plant parameters required to evaluate the status of the core cooling process. Furthermore, event data, status of safety-related systems and radiological data would also be transmitted for purposes of evaluating licensee's actions, safety of the plant, and radiological releases. In our opinion, the quantity of data to be transmitted by the link is a small fraction of the total data required to operate and control the plant.

The purpose of this letter is to advise you of the foregoing and to encourage your cooperation and participation to ensure timely completion of the evaluation of the NDL. It is also recognized that several related programs are also being implemented based upon the results of the NRR Lessons Learned Task Force Reports (NUREG-0578 and -0585); e.g., the Onsite Technical Support Center (OSTSC) and Control Room Evaluation. These actions will interface with the NDL and the OC. It is apparent that close coordination is needed between the requirements being established for the NDL and those of the OSTSC and therefore improved control rooms at individual plants.

To fully understand the plant parameter data systems presently in use at each nuclear plant, the staff through the Office of Nuclear Regulatory Research and the Sandia Laboratories will conduct a survey over the next six months. The results of this survey will then be used to finalize the specifications for an NDL such as digitized data access protocol and transmission link design requirements. The immediate objective will be to select lead plants and data to ensure early and meaningful operation of the NDL. A copy of the initial NDL specifications is enclosed for your information. The data protocol will be a fixed format data matrix that would be transmitted by all plants. Initially, blanks would be transmitted where data access is not available at a given plant. Subsequent actions by licensees would be to upgrade their capability to complete the data protocol and ensure uniform digitized computer data capability from each nuclear plant to the OC.

March 12, 1980

We will keep you advised of our progress in the NDL program. We are presently in contact with members of INPO who are helping in establishing basic requirements.

We appreciate your help and cooperation and your personal assistance in our deliberations to date and solicit your continued support in this important matter.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors

Enclosure:
Nuclear Data Link Specifications
February 21, 1980

APPENDIX H

Disturbance Analysis and Surveillance Systems

DISTURBANCE ANALYSIS AND SURVEILLANCE SYSTEMS

DASS is a computerized system to assist reactor operators in identifying the status of the plant and in taking acts to affect that status. A vast quantity of plant status and alarm information is first processed by the system to identify deviations which are significant given the mode of plant operations. If off-normal parameters are identified, then DASS uses a predefined model of the plant to integrate many pieces of information in order to rapidly determine the underlying problem and its possible consequences if left uncorrected. This concept is in contrast to the conventional approach of displaying and alarming each variable independently. There are two parallel DASS programs underway now.

EPRI/Westinghouse/Commonwealth Edison/Sargent & Lundy System Control

The general objectives of the EPRI, et al program is to:

1. Develop and evaluate improved process-operator interface in the following areas: real time information qualification, data verification, analyses, diagnoses, integration and prioritization.
2. Undertake phased implementation.
3. Build an adaptable, reliable, correct, unambiguous open loop system.

DOE/Sandia/Babcock/Wilcox/Burns & Roe/GPC

The general objectives of the DOE, et al program are to develop advanced monitoring and operator assistance techniques, safety margins, diagnostic information, recommend disturbance corrective procedures and improved transient control.

The DASS Advisory Group is seeking criteria within which longer term development could proceed. They want to study the cost benefit aspects. They do not want some unjust overcommitment to some new technique before it is cost feasible and beneficial.

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