NUREG/CR-1270
Vol. III

Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2

Final Report

Prepared by T. B. Malone, M. Kirkpatrick, K. Mallory, D. Eike, J. H. Johnson, R. W. Walker

The Essex Corporation

Prepared for
U. S. Nuclear Regulatory
Commission

Available from

GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

and

National Technical Information Service
Springfield, Virginia 22161

NUREG/CR-1270
Vol. III

Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2

Final Report

Manuscript Completed: December 1979
Date Published: January 1980

Prepared by
T. B. Malone, M. Kirkpatrick, K. Mallory, D. Eike, J. H. Johnson, R. W. Walker

The Essex Corporation
333 N. Fairfax Street
Alexandria, VA 22314

Prepared for
Three Mile Island Special Inquiry Group
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC Contract No. 04-79-209

TABLE OF CONTENTS

APPENDICES
Part 2

APPENDIX M

HUMAN ENGINEERING CRITERIA BEFORE 1973

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Industry Standards with Direct Human Factors Application

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE-603-O-1 IEE-603-I-1	Operator/System Integration Standard Instrumentation and Control Standard	4. Safety System Functional and Design Requirements, page 13. 4.2.1 Means shall be provided in the control room to implement manual initiation at the system level of the automatically initiated protective actions. The means provided shall minimize the number of discrete operator manipulations and shall depend on the operation of a minimum of equipment.	1968	56	Yes
IEE-603-I-4	Instrumentation and Control Standard	4. Safety System Functional and Design Requirements, page 13. 4.11 Information Displays 4.11.1 Displays for Protective Actions Initiated Solely by Manual Means. The display instrumentation provided for the manually initiated actions required for the safety system to accomplish its protective function shall be part of the safety system. The design shall minimize the possibility of anomalous indications which could be confusing to the operator.	1968	56	Yes
IEE-603-I-5	Instrumentation and Control Standard	4. Safety System Functional and Design Requirements, page 13. 4.11.2 System Status Indication. The display instrumentation provided for safety system status indication need not be part of the safety system. The display instrumentation shall provide accurate, complete, and timely information pertinent to safety system status. This information shall include indication and identification of protective actions at the channel level and the system level. The design shall minimize the possibility of anomalous indications which could be confusing to the operator.	1968	56	Yes
IEE-603-I-6	Instrumentation and Control Standard	4. Safety System Functional and Design Requirements, page 13. 4.11.3 Indication of Bypasses. If the protective actions of some part of the safety system have been bypassed or deliberately rendered inoperative for any purpose, continuing indication of this fact at the system level shall be provided in the control room.	1968	56	Yes

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Industry Standards with Direct Human Factors Application

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.1-I-3	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.2.4.6 Continuous indication of each control assembly position shall be provided in the control room.	1973		Yes
ANS 51.1-I-5 ANS51.1-O-1	Instrumentation and Control Standard Operator/System Integration Standard	5. Design Criteria, page 9. 5.3.4.3 Alarms shall be provided to alert the operator that process variables are approaching or have reached levels that initiate safety action. The alarm signals shall be obtained as close as practical to their source. Data presentation of these alarms shall be readily distinguished from other alarms. Acknowledgement of the alarm from one channel shall not inhibit the alarm of redundant channels.	1973		Yes
ANS51.1-O-2	Operator/System Integration Standard	5. Design Criteria, page 9. 5.3.4.4 The data displayed and controls located in the control room shall be adequate: (1) to regulate the process variables within their normal limits (2) to cope with malfunctions or accidents (3) to assess accidents and perform necessary actions for recovery.	1973		Yes
ANS3.2-P-5	Operator Procedure Standard	5. Program, Policies and Procedures, page 8. 5.3.2 Procedure Content. The format of procedures may vary from plant to plant, depending on the policies of the owner organization. However, procedures shall include, as appropriate, the following elements: (1) Title (2) Statement of Applicability (3) References (4) Prerequisites (5) Precautions (6) Limitations and Actions (7) Main Body (8) Acceptance Criteria (9) Checkoff Lists	1976		Yes

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED CRITERIA AND GUIDES

Reference: U.S. Regulatory Guides With Direct Human Factors Application

Number	Type of Criterion or Guide	Language of Criterion or Guide	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
RG1.70-MG-6	Policy, Planning and Management Guide Instrumentation and Control Guide	RG 1.70 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, Revision 3, November 1978, page 9-9. 9.3.5.5 Instrumentation Requirements. The system instrumentation and controls should be described. The adequacy of safety-related instrumentation and controls to fulfill their functions should be demonstrated.	1972	66	Yes
RG1.70-MG-7	Policy, Planning and Management Guide Instrumentation and Control Guide Operator/System Integration Guide Control Room Environment Guide	RG 1.70 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, Revision 3, November 1978, page 9-9. 9.4.1.1 Design Bases. The design bases for the air treatment system for the control room and other auxiliary rooms (e.g., relay rooms and emergency switchgear rooms) considered to be part of the control areas should be provided. Include the design criteria (e.g., single failure), requirements for the manual or automatic actuation of system components or isolation dampers, ambient temperature and humidity requirements, criteria for plant operator comfort and safety, requirements for radiation protection and monitoring of abnormal radiation levels and other airborne contaminants, and environmental design requirements.	1972	66	Yes
RG1.70-MG-10	Policy, Planning and Management Guide Control Room Environment Guide	RG 1.70 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, Revision 3, November 1978, page 9-19. 9.5.3 Lighting Systems A description of the normal lighting system for the plant should be provided. A description of the emergency lighting system, including design criteria and a failure analysis, should also be provided.	1972	66	Yes

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED CRITERIA AND GUIDES

Reference: U.S. Regulatory Guides With Direct Human Factors Application

Number	Type of Criterion or Guide	Language of Criterion or Guide	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
RG1.62-IG-1	Instrumentation and Control Guide Operator/System Integration Guide	RG 1.62 Manual Initiation of Protective Actions, October 1973, page 1. 1. Means should be provided for manual initiation of each protective action (e.g., reactor trip, containment isolation) at the system level, regardless of whether means are also provided to initiate the protective action at the component or channel level (e.g., individual control rod, individual isolation valve).	1973		Yes
RG1.62-IG-3	Instrumentation and Control Guide	RG 1.62 Manual Initiation of Protective Actions, October 1973, page 1. 3. The switches for manual initiation of protective actions at the system level should be located in the control room and be easily accessible to the operator so that action can be taken in an expeditious manner.	1973		Yes
RG1.47-IG-1	Instrumentation and Control Guide Operator/System Integration Guide	RG 1.47 Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems, May 1973, page 2. Bypass indication should aid the operator in recognizing the effects on plant safety of seemingly unrelated or insignificant events. Therefore, the indication of bypass conditions should be at the system level, whether or not it is also at the component or channel level.	1973		Yes
RG1.47-IG-5	Instrumentation and Control Guide	RG 1.47 Bypassed and Inoperable Status Indications for Nuclear Power Plant Safety Systems, May 1973, page 3. 4. Manual capability should exist in the control room to activate each system-level indicator provided in accordance with C.1. above.	1973		Yes

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED CRITERIA AND GUIDES

Reference: U.S. Regulatory Guides With Direct Human Factors Application

Number	Type of Criterion or Guide	Language of Criterion or Guide	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
RG1.70-PG-1	Operator Procedure Guide Human Factors Test and Evaluation Guide Policy, Planning and Management Guide	RG 1.70 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, Revision 3, November 1978, page 9-19. 9.5.2.3 Inspection and Testing Requirements. The inspection and testing requirements for the communication systems should be provided.	1972	66	Yes
SG11-IC-1	Instrumentation and Control Criterion	SG 11 Instrument Lines Penetrating Primary Reactor Containment, 3/10/71, page 2. The status (opened or closed) of all such isolation valves should be indicated in the control room. If a remotely operable valve is provided, sufficient information should be available in the control room or other appropriate location to assure timely and proper actions by the operator.	1971		Yes

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

APPENDIX N

INDUSTRY STANDARDS CRITERIA

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations, ANSI/ANS-4.1, 1978.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS4.1-O-1	Operator/System Integration Standard	3. Design and Basic Requirements, page 7. The inadvertent initiation and execution of a protective action shall not result in damage to any fission product barrier or safety system which is inconsistent with the limiting safety consequences of the category of events to which such inadvertent action belongs.	1978		
ANS4.1-O-2 ANS4.1-P-1	Operator/System Integration Standard Operator Procedure Standard	3. Design and Basic Requirements, page 7. 3.6.6 Operator Participation. The safety systems shall be capable of performing the protective functions without requiring the reactor operator to take any action prior to a defined time limit following each Design Basis Event. After the time limit, operator participation may be used to maintain safe conditions. This time limit shall be appropriate for the actions required, the number and location of operators, the information available to the operator, and the number and location of controls, and any design features provided to protect the operator.[6]	1978		
ANS4.1-O-3 ANS4.1-P-2 ANS4.1-I-1	Operator/System Integration Standard Operator Procedure Standard Instrumentation and Control Standard	3. Design and Basic Requirements, page 8. The designers shall determine, by means of a systematic analysis, that (a) the monitored process variable can provide the required information during the Design Basis Events. (b) the equipment can perform in the configuration specified for its installation. (c) the interactions of protective actions, control actions, and the environmental changes that caused, or are caused by, the Design Basis Events do not prevent the mitigation of the consequences of the event; and	1978		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations, ANSI/ANS-4.1, 1978.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
		3. Design and Basic Requirements, page 8 (continued) (d) the equipment in the configuration specified for its installation cannot easily be made inoperational by the inadvertent actions of operating or maintenance personnel.			
ANS4.1-O-4 ANS4.1-P-3	Operator/System Integration Standard Operator Procedure Standard	3. Design and Basic Requirements, page 8. 3.8 <u>Operation and Maintenance</u>. The design of the safety systems and the safety supporting systems shall permit implementation of operating and maintenance procedures for the surveillance, calibration, adjustment, and repair of the protection and actuator systems without inducing a Design Basis Event or an unprotected condition. The designer shall give special consideration to preventing inadvertent modification of the systems that may negate the intent of the system design.	1978		
ANS4.1-I-2	Instrumentation and Control Standard	3. Design and Basic Requirements, page 9. 3.9 <u>Surveillance</u>. Means for surveillance of the safety systems and the safety supporting systems shall be established. They shall be adequate to: (a) determine that the performance of the safety systems and their safety supporting systems is within prescribed limits; (b) assure that maintenance operations have been performed correctly; (c) detect trends toward unacceptable conditions; and (d) determine that the independence of redundant or diverse systems has been maintained. (e) permit the operational capability of an instrument channel, logic channel, and an actuator channel to be demonstrated.	1978		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Gaseous Radioactive Waste Processing Systems for Light Water Reactor Plants, ANSI/ANS 55.4, 1979.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS55.4-I-1	Instrumentation and Control Standard	7. Instrumentation and Controls, page 12. 7.2 PWR Instrumentation and Controls. The PWR Gaseous Radioactive Waste Processing System shall have sufficient instrumentation and controls such that it can be started, operated, monitored and shutdown from a remote control area, located in radiation Zone I or II (see Table 7). Positive operator action shall be required to effect any controlled discharge to the environment.	1977	49	
ANS55.4-I-2	Instrumentation and Control Standard	7. Instrumentation and Controls, page 12. 7.3 Process and Effluent Radiation Monitoring. The effluent radiation monitoring devices shall be designed to continuously monitor and record all gaseous radioactivity released from the BWR Main Condenser Offgas System and PWR Gaseous Radioactive Waste Processing System to the atmosphere through normal release pathways. Effluent radiation monitors in the systems shall automatically terminate release upon high radiation (above a predetermined set point) in the discharge. Monitor readout shall be in the main control room. Additional monitor readout may be provided in a central control area to facilitate system control.	1977	49	
ANS55.4-IR-1	Instrumentation and Control Recommended Practice	7. Instrumentation and Controls, page 12. Table 6 gives the minimum requirements for instrumentation and controls. In addition it gives specific recommendations which will provide information and control features for the following purposes during startup, operation and shutdown of the system: (1) Provide information on hydrogen concentration or oxygen concentration, or both. (2) Provide information on system or component pressures to protect against over-pressurization and to enable proper flow. (3) Provide information on liquid accumulation in tanks so that drainage can be accomplished when required. (4) Provide information on cooling water, oil, air and other service systems to insure that components are operating properly and to enable identification of malfunctions. (5) Provide information such as inlet and outlet temperatures of process gas in heat exchangers, liquid level in gas condensers, moisture content from gas conditioning equipment and adsorber vault temperature to facilitate equipment performance evaluation and allow corrective measures to be taken when required.	1977	49	

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Gaseous Radioactive Waste Processing Systems for Light Water Reactor Plants, ANSI/ANS 55.4, 1979.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
		7. Instrumentation and Controls, page 12 (continued). (6) Provide information on recombiner performance. (7) Provide discharge flow rate information to enable adequate dispersion and determination of radioactivity release rates. (8) Provide informaton on radioactivity concentrations to determine atmospheric release rates, holdup times and equipment performance. Also to provide for the automatic termination of releases to the atmosphere when necessary. Valve(s) used for automatic termination of release shall be designed to fail-closed in the event that power is lost to the valve(s).			

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation, ANSI N320, 1979.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
AIN320-I-1	Instrumentation and Control Standard	4. General Consideration for Emergency Instrumentation, page 8. Primary emphasis is placed on the selection of instruments and instrument systems and on their ability to provide data rapidly as basis for making appropriate emergency action decisions. The instrumentation should include both installed systems, herein referred to as systems, with appropriate readouts and portable instruments, since either portable or installed instrumentation alone may provide incomplete information.	1979		
AIN320-I-2 AIN320-E-1	Instrumentation and Control Standard Operator Support Equipment Standard	4. General Consideration for Emergency Instrumentation, page 8. (1) Installed instrumentation systems with remote readout to a safe location capable of characterizing releases to containment and auxiliary buildings and the radiological problems associated with evacuation and reentry. These systems should be provided with a remote readout at a location which will be habitable under accident conditions.	1979		
AIN320-E-2	Operator Support Equipment Standard	4. General Consideration for Emergency Instrumentation, page 8. (2) Portable survey instruments to supplement installed instrument systems to permit estimation of exposure to persons, to locate radiation sources and determine their distribution, and to make radiological measurements that may become of ad hoc interest at locations not covered by installed instrumentation. In determining the type of instrumentation required, the following apply: 4.1.1 Continuous measurement of airborne radioactivity in the containment is necessary. 4.1.2 Where appropriate, air sampling systems shall be consistent with the requirements stated in ANSI N13.1-1969, American National Standard Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities. 4.1.3 Remote area monitoring systems are necessary for measuring the ambient radiation field at points within the reactor facility. The system should be capable of measurement over a wide spectrum of energies and range of exposure rates. 4.1.4 High range monitoring systems are necessary for assessment of effluent radioactive material.	1979		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation, ANSI N320, 1979.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
		4. General Consideration for Emergency Instrumentation, page 8 (continued). 4.1.5 High range portable survey instruments and personnel dosimeters are necessary to permit rapid assessment of high exposure rates and time-integrated dose. 4.1.6 Instrumentation should be capable of performing as intended, considering the total environment to which the instrumentation will be exposed during emergencies. Physical protection is usually necessary.			
AIN320-I-3	Instrumentation and Control Standard	5. Criteria for Radiological Instrumentation Systems, page 9. 5.1 A normally active internal audit circuit which tests both the detector and electronics shall be provided and shall present an appropriate signal at a centrally manned location in the event of a malfunction or failure.	1979		
AIN320-I-4	Instrumentation and Control Standard	5. Criteria for Radiological Instrumentation Systems, page 9. 5.5 Switches and other controls shall be protected to avoid inadvertent deactivation or inadvertent maloperation of system.	1979 --		
AIN320-I-5	Instrumentation and Control Standard	5. Criteria for Radiological Instrumentation Systems, page 9. 5.6 The ranges of emergency instrumentation systems should overlap the ranges of instrumentation systems for routine or nonemergency monitoring. (The minimum ranges specified herein generally assume a one decade overlap.)	1979		
AIN320-I-6	Instrumentation and Control Standard	5. Criteria for Radiological Instrumentation Systems, page 9. 5.7 Overall system accuracy (does not include sample accuracy) shall be within ± 40 percent at the 95 percent confidence level over the entire operating range, with precision within ± 15 percent for any single measurement level.	1979		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation, ANSI N320, 1979.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
AIN320-I-7	Instrumentation and Control Standard	5. Criteria for Radiological Instrumentation Systems, page 9. 5.15 Logarithmic, quasilogarithmic, or digital readout scales should be considered. If multiple scales are used, automatic range changing shall be provided and the range that is in operation shall be clearly displayed.	1979		
AIN320-I-8	Instrumentation and Control Standard	5. Criteria for Radiological Instrumentation Systems, page 9. 5.16 Readout capability and alarms shall be provided in the control room. Readout and alarms should also be provided at or near the detector.	1979		
AIN320-I-9	Instrumentation and Control Standard	5. Criteria for Radiological Instrumentation Systems, page 9. 5.17 All units of similar function, including detectors, electronic modules, readout and display devices and power supplies, should be interchangeable. Operable spare units shall be available.	1979		
AIN320-I-10	Instrumentation and Control Standard	5. Criteria for Radiological Instrumentation Systems, page 9. 5.18 The units of the system should be capable of being functionally tested without removal from the instrument system.	1979		
AIN320-I-11 AIN320-O-1	Instrumentation and Control Standard Operator/System Integration Standard	5. Criteria for Radiological Instrumentation Systems, page 9. 5.19 Instrument systems shall be equipped with alarms capable of being externally set to alarm at any selected point within the stated range and shall continue to operate above the selected alarm points. Audible alarms shall be incapable of reset without active acknowledgements. Such acknowledgements shall retain the visual alarm until the signal is below the alarm setting. If the audible is not acknowledged, decrease of the signal below the trip setting shall not reset the visual alarm.	1979		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, ANSI N18.7/ANS3.2, 1976.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS3.2-M-1	Policy, Planning and Management Standard	4. Reviews and Audits, page 5. 4.1 General. Programs for reviews and for audits of activities affecting plant safety during the operational phase shall be established by the owner organization to:	1976		
ANS3.2-M-2	Policy, Planning and Management Standard	4. Reviews and Audits, page 5. (3) Verify that reportable events, which require reporting to NRC in writing within 24 hours, are promptly investigated and corrected in manner which reduces the probability of recurrence of such events.	1976		
ANS3.2-M-3	Policy, Planning and Management Standard	5. Program, Policies and Procedures, page 8. 5.2.1 Responsibilities and Authorities of Operating Personnel. The responsibilities and authorities of the plant operating personnel shall be delineated. These shall include, as a minimum: (1) The reactor operator's authority and responsibility for shutting the reactor down when he determines that the safety of the reactor is in jeopardy or when operating parameters exceed any of the reactor protection system set-points and automatic shutdown does not occur. (2) The responsibility to determine the circumstances, analyze the cause, and determine that operations can proceed safely before the reactor is returned to power after a trip or an unscheduled or unexplained power reduction. (3) The senior reactor operator's responsibility to be present at the plant and to provide direction for returning the reactor to power following a trip or an unscheduled or unexplained power reduction. (4) The responsibility to believe and respond conservatively to instrument indications unless they are proved to be incorrect. (5) The responsibility to adhere to the plant's Technical Specifications. (6) The responsibility to review routine operating data to assure safe operation.	1976		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, ANSIN18.7/ANS3.2, 1976.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS3.2-M-4 ANS3.2-P-1	Policy, Planning and Management Standard Operator Procedure Standard	5. Programs, Policies and Procedures, page 8. 5.2.2 Procedure Adherence. Procedures shall be followed, and the requirements for use of procedures shall be prescribed in writing. Rules shall be established which provide methods by which temporary changes to approved procedures can be made, including the designation of a person or persons authorized to approve such changes. Temporary changes which clearly do not change the intent of the approved procedure, shall as a minimum be approved by two members of the plant staff knowledgeable in the areas affected by the procedures. At least one of these individuals shall be the supervisor in charge of the shift and hold a senior operators license on the unit affected. Such changes shall be documented and, if appropriate, incorporated in the next revision of the affected procedure. In the event of an emergency not covered by an approved procedure, operations personnel shall be instructed to take action so as to minimize personnel injury and damage to the facility and to protect health and safety. Guidance should be provided to identify the manner in which procedures are to be implemented. Examples of such guidance include identification of those tasks that require: (1) The written procedure to be present and followed step by step while the task is being performed (2) The operator to have committed the procedural steps to memory (3) Verification of completion of significant steps, by initials or signatures of checkoff lists. The types of procedures that shall be present and referred to directly are those developed for extensive or complex jobs where reliance on memory cannot be trusted, e.g., reactor start-up, tasks which are infrequently performed, and tasks in which operations must be performed in a specified sequence. Procedural steps for which actions should be committed to memory include, for example, immediate actions in emergency procedures. Routine procedural actions that are frequently repeated may not require the procedure to be present. Copies of all procedures shall be available to appropriate members of the plant staff. If documentation of an action is required, the necessary data shall be recorded as the task is performed. Examples of procedures requiring verification are furnished in 5.3.4.1 and 5.3.4.2	1976		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, ANSIN18.7/ANS3.2, 1976.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS3.2-M-5	Policy, Planning and Management Standard	5. Program, Policies and Procedures, page 8. 5.2.3 Operating Orders. A mechanism shall be provided for dissemination to the plant staff of instructions of general and continuing applicability to the conduct of business. Such instructions, sometimes also referred to as standing orders or standard operating procedures, should deal with job turnover and relief, designation of confines of control room, definition of duties of operators and others, transmittal of operating data to management, filing of charts, limitations on access to certain areas and equipment, shipping and receiving instructions, or other such matters. Provisions should be made for periodic review and updating of standing orders.	1976		
ANS3.2-M-6 ANS3.2-P-2	Policy, Planning and Management Standard Operator Procedure Standard	5. Program, Policies and Procedures, page 8. 5.2.4 Special Orders. A mechanism shall be provided for issuing mangement instructions which have short-term applicability and which require dissemination. Such instructions, sometimes referred to as a special orders, should encompass special operations, housekeeping, data taking, publications and their distribution, plotting process parameters, personnel actions, or other similar matters. Provisions should be made for periodic review, updating and cancellation of special orders.	1976		
ANS3.2-M-7 ANS3.2-P-3	Policy, Planning and Management Standard Operator Procedure Standard	5. Program, Policies and Procedures, page 8. 5.2.5 Temporary Procedures. Temporary procedures may be issued during the operational phase: to direct operations during testing, refueling, maintenance and modifications; to provide guidance in unusual situations not within the scope of the normal procedures; and to insure orderly and uniform operations for short periods when the plant, a system, or a component of a system is performing in a manner not covered by existing detailed procedures or has been modified or extended in such a manner that portions of existing procedures do not apply. Temporary procedures shall include designation of the period of time during which they may be used and shall be subject to the review process prescribed in 4.3 and 5.2.15 as applicable. Temporary procedures shall be approved by the management representative assigned approval authority.	1976		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, ANSIN18.7/ANS3.2, 1976.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS3.2-M-8 ANS3.2-P-4	Policy, Planning and Management Standard Operator Procedure Standard	5. Program, Policies and Procedures, page 8. 5.2.6 Equipment Control. Permission to release equipment or systems for maintenance shall be granted by designated operating personnel. Prior to granting permission, such operating personnel shall verify that the equipment or system can be released, and determine how long it may be out of service. Granting of such permission shall be documented. Attention shall be given to the potentially degraded degree of protection when one subsystem of a redundant safety system has been removed for maintenance.	1976		
ANS3.2-H-1	Human Factors Test and Evaluation Standard	5. Program, Policies and Procedures, page 8. (1) Tests during the preoperational period to demonstrate that performance of plant systems is in accordance with design intent and that the coordinated operation of the plant as a whole is satisfactory, to the extent feasible.	1976		
ANS3.2-H-2	Human Factors Test and Evaluation Standard	5. Program, Policies and Procedures, page 8. (2) Tests during the initial operational phase to demonstrate the performance of systems that could not be tested prior to operation and to confirm those physical parameters, hydraulic or mechanical characteristics that need to be known, but which could not be predicted with the required accuracy, and to confirm that plant behavior conforms to design criteria.	1976		
ANS3.2-P-5	Operator Procedure Standard	5. Program, Policies and Procedures, page 8. 5.3.2 Procedure Content. The format of procedures may vary from plant to plant, depending on the policies of the owner organization. However, procedures shall include, as appropriate, the following elements: (1) Title (2) Statement of Applicability (3) References (4) Prerequisites (5) Precautions (6) Limitations and Actions (7) Main Body (8) Acceptance Criteria (9) Checkoff Lists	1976		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, ANSIN18.7/ANS3.2, 1976.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS3.2-P-5 (continued)	Operator Procedure Standard	5. Program, Policies and Procedures, page 8. (continued) 5.3.3 System Procedures 5.3.4 General Plant Procedures 5.3.4.1 Startup Procedures (1) Prerequisites (2) Main Body 5.3.4.2 Shutdown Procedures 5.3.4.3 Power Operation and Load Changing Procedures 5.3.4.4 Process Monitoring Procedures 5.3.4.5 Fuel-Handling Procedures (1) Prerequisites (2) Main Body 5.3.5 Maintenance Procedures (1) Preparation for Maintenance (2) Performance of Maintenance (3) Post Maintenance Check Out and Return to Service (4) Supporting Maintenance Documents 5.3.6 Radiation Control Procedures 5.3.7 Calibration and Test Procedures 5.3.8 Chemical-Radiochemical Control Procedures 5.3.9 Emergency Procedures 5.3.9.1 Emergency Procedure Format and Content (1) Title (2) Symptoms (3) Automatic Actions (4) Immediate Operator Actions (5) Subsequent Operator Actions 5.3.9.2 Events of Potential Emergency 5.3.9.3 Procedures for Implementing Emergency Plans 5.3.10 Test and Inspection Procedures	1976		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Containment Isolation Provisions for Fluid System ANS 56.2, 1976.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS56.2-I-1	Instrumentation and Control Standard	4. Design Requirement, page 9. 4.2.2 All power-operated isolation valves shall be capable of remote manual actuation from the main control room.	1974	50	
ANS56.2-I-2 ANS56.2-O-1	Instrumentation and Control Standard Operator/System Integration Standard	4. Design Requirement, page 9. 4.2.3 All power-operated isolation valves shall have provisions in the control room for indication of the status of the valve showing open and closed positions. A failure of an indication circuit should not cause a failure of the actuation circuit. All electric power-operated isolation valves shall have provisions in the control room for indication of the availability of power at the line side of the motor starter, e.g., position indicating lights energized from control power transformer. Sealed closed isolation valves are under administrative controls and do not require position indication in the control room for valve status.	1974	50	
ANS56.2-I-3 ANS56.2-O-2	Instrumentation and Control Standard Operator/System Integration Standard	4. Design Requirement, page 9. For power-operated isolation valves which automatically operate upon receipt of a containment isolation signal, the automatic initiating signal shall be the primary mode and the secondary mode shall be a remote manual initiation from the main control room. It should not be possible for remote manual operation to override the automatic isolation signal until the sequence of automatic events following an isolation signal is completed. The design of the override shall necessitate a deliberate, premeditated action on the part of the operator (e.g., key interlocked switch or manual "hold-open" with return to automatic closure.)	1974	50	
ANS56.2-I-4	Instrumentation and Control Standard	4. Design Requirement, page 9. For power-operated isolation valves which do not receive a containment isolation signal, the primary mode shall be a remote manual initiation signal from the main control room. Those valves outside the containment should have a local secondary mode of operation, e.g., handwheel. Those valves inside containment need not have a secondary mode of operation.	1974	50	

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Containment Isolation Provisions for Fluid System ANS 56.2, 1976.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS56.2-O-3	Operator/System Integration Standard	4. Design Requirement, page 9. 4.2.4 Isolation valve closure shall be completed when a isolation signal is received and the valve shall not be opened until the signal is removed and deliberate operator action is taken (reset switch). This is to prevent the valve from returning to the pre-accident condition automatically when the isolation signal is removed.	1974	50	
ANS56.2-O-4	Operator/System Integration Standard	4. Design Requirement, page 9. 4.12 <u>Determination of Isolation Requirements for Remote Manual Controlled Systems.</u> Remote manual valves may be provided on engineered safety features or engineered safety feature related systems in order to maintain containment or preserve system function in the event of a leak or line break in such systems. Provisions shall be made to allow the main control room operator to know when to isolate the affected line. An analysis of the consequences of a leak or line break in these systems shall be made in order to determine how fast the operator shall isolate the line. The results of this analysis shall be used to determine the provisions needed to alert the operator that the line requires isolation. The provisions which indicate the requirement for isolation may include devices which measure parameters such as flow, temperature, pressure, noise, radiation, and sump water level outside containment.	1974	50	
ANS56.2-I-5	Instrumentation and Control Standard	5. Testing, page 13. Control switches, limit switches, visual accessibility, indicating lights, fluid system characteristics, indicators, etc., as necessary, shall be provided to permit valve exercising testing.	1974	50	

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANS 51.1, 1973.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.1-I-1	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.2.4.2 Sources of reactor spatial instability shall be examined and the design shall be such that one of the following conditions is applicable:	1973		
ANS51.1-I-2	Instrumentation and Control Standard	5. Design Criteria, page 9. (3) a control system with appropriate means for detection is provided that is capable of limiting the instability to within core structural design limits.	1973		
ANS51.1-I-3	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.2.4.6 Continuous indication of each control assembly position shall be provided in the control room.	1973		
ANS51.1-I-4	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.3.4.2 In addition to information readouts required by N42.7-1972 (14) (see 5.3.4.1), information pertinent to the monitoring of each safety process variable shall be available to the reactor operator.	1973		
ANS 51.1-I-5 ANS51.1-O-1	Instrumentation and Control Standard Operator/System Integration Standard	5. Design Criteria, page 9. 5.3.4.3 Alarms shall be provided to alert the operator that process variables are approaching or have reached levels that initiate safety action. The alarm signals shall be obtained as close as practical to their source. Data presentation of these alarms shall be readily distinguished from other alarms. Acknowledgement of the alarm from one channel shall not inhibit the alarm of redundant channels.	1973		
ANS51.1-I-6 ANS51.1-C-1	Instrumentation and Control Standard Control Room Environment Standard	5. Design Criteria, page 9. 5.3.4.5 Adequate data displays and controls shall be provided outside the control room to shut down and maintain the reactor in a safe "Hot Standby" condition in the event the control room becomes uninhabitable.	1973		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANS 51.1, 1973.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.1-O-2	Operator/System Integration Standard	5. Design Criteria, page 9. 5.3.4.4 The data displayed and controls located in the control room shall be adequate: (1) to regulate the process variables within their normal limits (2) to cope with malfunctions or accidents (3) to assess accidents and perform necessary actions for recovery.	1973		
ANS51.1-I-7	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.3.3.10 Instrumentation shall be provided in the reactor coolant pressure boundary to demonstrate that core power and system temperatures, pressures, flows, and coolant volumes are maintained within safety limits prescribed for the design.	1973		
ANS51.1-I-8	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.3.3.16 Means shall be provided for detecting and measuring leakage from the reactor coolant pressure boundary.	1973		
ANS51.1-I-9	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.3.3.17 For the reactor coolant pressure boundary, the following shall be displayed or alarmed in the control room, or both: (1) pressurizer or reactor coolant pressure boundary pressure (2) pressurizer liquid level (3) system temperatures (4) coolant flow rates (5) principal parameters affecting the reactor coolant pump motor assembly operation (6) status indication of power-operated valves.	1973		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANS 51.1, 1973.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.1-I-14	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.5.3.9 Instrumentation provided for safety system functions of the reactor coolant auxiliary systems shall be in accordance with applicable criteria of 5.3.	1973		
ANS51.1-I-15	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.5.3.12 For the reactor coolant auxiliary systems, the following shall be displayed or alarmed in the control room, or both: (1) coolant letdown flow (2) coolant makeup flow (3) flow of demineralized makeup (4) flow of boric acid makeup (5) letdown stream pressure (6) surge tank gas pressure (7) temperature of letdown stream (heat exchanger outlet) (8) temperature of surge tank discharge stream (9) temperature of discharge from regenerative heat exchanger entering reactor coolant system (10) liquid level of surge tank (11) liquid level of boric acid tank(s) (12) status indication of principal pumps (13) status indication of power-operated valves.	1973		
ANS51.1-I-16	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.6.3.5 Instrumentation shall be provided as required to demonstrate that component and process cooling systems performance objectives are met and systems temperatures and pressures are controlled within safety limits prescribed for the designs.	1973		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANS 51.1, 1973.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.1-I-10	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.4.3.9 Instrumentation shall be provided as required to demonstrate that residual heat removal system performance objectives are met and system temperatures and pressures are controlled within safety limits prescribed for the design.	1973		
ANS51.1-I-11	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.4.3.11 Instrumentation provided for the safety system functions of the residual heat removal system shall be in accordance with applicable criteria of 5.3.	1973		
ANS51.1-I-12	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.4.3.13 For the residual heat removal system, the following shall be displayed or alarmed in the control room, or both: (1) system pressure (2) reactor coolant flow rate through the system (3) system temperatures (4) status indication of pumps (5) status indication of power-operated valves.	1973		
ANS51.1-I-13	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.5.3.8 Instrumentation shall be provided as required to demonstrate that reactor coolant auxiliary systems performance objectives are met and systems temperatures and pressures are controlled within safety limits prescribed for the designs.	1973		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANS 51.1, 1973.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.1-1-17	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.6.3.8 For those portions of the service water system performing safety functions the following shall be displayed or alarmed in the control room, or both: (1) flow rates for cooling coil supplies of the air cooling subsystem, if used as an engineered safety feature (2) radioactivity of service water from potentially high level sources (3) status indication of pump (4) status indication of power-operated valves.	1973		
ANS51.1-1-18	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.6.3.9 For the intermediate cooling water system, the following shall be displayed or alarmed in the control room, or both: (1) temperature of water supply to principal system heat exchangers (2) surge tank liquid level (3) radioactivity level in system (4) status indication of pumps (5) status indication of power-operated valves.	1973		
ANS51.1-1-19	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.7.3.9 Instrumentation shall be provided as required to demonstrate that secondary system performance objectives are met and system temperatures and pressures are controlled within safety limits prescribed for the design.	1973		
ANS51.1-1-20	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.7.3.11 Design shall provide means to detect potential radioactivity in secondary system coolant.	1973		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANS 51.1, 1973.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.1-I-21	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.4.7.3.12 For the secondary system, the following shall be displayed or alarmed in the control room, or both: (1) feedwater flow rate (normal and emergency) (2) steam pressure (3) feed header pressure (each steam generator) (4) emergency feed pump discharge pressure (5) each steam generator liquid level (6) condensate storage tank liquid level (7) radioactivity (at air ejector discharge and steam generator blow-down points) (8) status indication of emergency feed pumps (9) status indication of power-operated valves.	1973		
ANS51.1-I-22	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.3.3.10 All power-operated valves required for reactor containment isolation shall be capable of remote actuation on signal from the main control room.	1973		
ANS51.1-I-23	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.3.3.12 All power-operated isolation valves of the reactor containment system shall be provided with remote position indication in the control room and such indication shall be independent of the closing signal or closing power device, or both.	1973		
ANS51.1-I-24	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.3.3.13 Instrumentation and controls for that portion of the reactor containment isolation system relied on to function under accident conditions shall be in accordance with the applicable criteria in 5.3.	1973		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANS 51.1, 1973.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.1-I-25	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.3.3.15 Instrumentation shall be provided for monitoring the reactor containment atmosphere for gaseous and particulate radioactivity. Readout of the same shall be provided in the control room.	1973		
ANS51.1-I-26	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.3.3.16 Visual indication shall be provided in the control room to indicate the open and closed status of the personnel air-lock doors.	1973		
ANS51.1-I-27	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.3.3.19 For the reactor containment system, the following shall be displayed or alarmed in the control room, or both: (1) internal pressure (2) internal temperature (3) internal humidity (4) reactor containment structure sump liquid level (5) radiation and radioactivity levels (6) status indication of power-operated valves, ventilation dampers and access openings relied upon for reactor containment isolation.	1973		
ANS51.1-I-28	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.4.3.10 All power-operated components required for the emergency core cooling system shall be capable of remote manual operation on signal from the control room.	1973		
ANS51.1-I-29	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.4.3.14 For the emergency core cooling system, the following shall be displayed or alarmed in the control room, or both: (1) emergency core cooling system flow (2) accumulator tank pressures (3) recirculated water temperature (4) accumulator tank liquid levels (5) refueling water storage tank (systems head tank) liquid level (6) status indication of pumps (7) status indication of power-operated isolation or transfer valves (independent of operating signal or operating power device, or both).	1973		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANS 51.1, 1973.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.1-1-30	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.5.3.11 All power-operated components required for the reactor containment cooling system shall be capable of remote manual operation on signal from the control room.	1973		
ANS51.1-1-31	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.5.3.13 Instrumentation and controls for that portion of the reactor containment cooling system relied on to function under accident conditions shall be in accordance with the applicable criteria of 5.3.	1973		
ANS51.1-1-32	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.5.3.15 For the reactor containment cooling system, the following shall be displayed or alarmed in the control room, or both: (1) spray system flow (2) status indication of spray pumps and air cooling blowers (3) status indication of power-operated dampers and control valves (independent of operating signal or operating power device, or both).	1973 -		
ANS51.1-1-23	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.6.3.10 All power-operated components required for the air cleanup system shall be capable of remote manual operation on signal from the control room.	1973		
ANS51.1-1-34	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.5.6.3.14 For the air cleanup system, the following shall be displayed or alarmed in the control room, or both: (1) flow from the additive tank (if used for spray system) (2) liquid level of additive tank (if used for spray system) (3) temperatures of filter beds (if charcoal filter used) (4) status indication of power-operated dampers and valves (independent of operating signal or operating power device, or both).	1973		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANS 51.1, 1973.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.1-I-35	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.6.4.20 Instrumentation necessary to monitor performance of the radioactive waste disposal system and provide for system control to maintain this performance shall be provided. The following parameters shall be measured: (1) activity level in waste gas discharge line (2) activity level in liquid waste discharge line (3) flow rate in waste gas discharge line (4) flow rate in liquid waste discharge line (5) pressure in headers and pressure vessels designed to contain radioactive waste gas above atmospheric pressure (6) - liquid level in liquid waste storage tanks.	1973		
ANS51.1-I-36	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.6.4.21 Instrumentation and radiation monitoring equipment and its means for periodic calibration shall be provided to monitor liquid and gaseous effluent discharged to the environs from the radioactive waste disposal system. This instrumentation shall be of a sensitivity sufficient to establish that the requirements of appropriate federal regulations for off-site radiation doses are not exceeded. Means shall be provided such that integrated quantity discharges of radioactivity can be determined.	1973		
ANS51.1-I-37	Instrumentation and Control Standard	5. Design Criteria, page 9. 5.6.4.24 Gaseous and liquid radioactive waste discharge lines of the radioactive waste disposal system shall be equipped with a shutoff valve that is automatically closed if a radiation monitor on that discharge line indicates the release of excessive amounts of radioactivity. The high monitor indication shall be alarmed at the radioactive waste control station and main control room.	1973		
ANS51.1-O-3	Operator/System Integration Standard	5. Design Criteria, page 9. 5.6.4.22 Positive operator action shall be required to effect any controlled discharge to environment. As a normal method, controlled discharges from storage tanks shall not be accomplished by gravity or siphoning flow.	1973		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems, ANSI/IEEE Std. 338, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE-338-O-1	Operator/System Integration Standard	4. Basis, page 8. Interrelationship among the systems, components, and human factors in each phase of the test activity shall be considered and reflected in the system design and layout.	1975	51	
IEE-338-I-1	Instrumentation and Control Standard	4. Basis, page 8. Provision shall be made for locating test equipment and access to test points to minimize the effort and time required to perform checks, inspections, functional tests, and calibration verification tests.	1975	51	
IEE-338-P-1	Operator Procedure Standard	4. Basis, page 8. Testing programs shall be conducted in a logical sequence such that the overall condition of the systems under test can immediately be assessed and the need for progressing further into the testing of individual components be determined.	1975	51	
IEE-338-P-2	Operator Procedure Standard	5. Design, page 8. The safety systems shall be designed to be testable during operation of the nuclear power generating station as well as during those intervals when the station is shut down. This testability shall permit the independent testing of redundant channels and load groups while (1) maintaining the capability of these systems to respond to bona fide signals during operation, (2) tripping the output of the channel being tested, or (3) bypass the equipment consistent with availability requirements.	1975	51	
IEE-338-I-2	Instrumentation and Control Standard	5. Design, page 8. (7) Each test bypass condition utilized at a frequency of more than once a year shall be individually and automatically indicated to the operator in the main control room in such a manner that the bypassing of a protective function is immediately evident and continuously indicated.	1975	51	

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems, ANSI/IEEE Std. 338, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE-338-E-1	Operator Support Equipment Standard	5. Design, page 8. (8) A means of communication shall be provided between remote testing stations and the main control room to ensure that station operators are cognizant of the status of those systems under test.	1975	51	
IEE-338-P-3	Operator Procedure Standard	6. Testing Program, page 9. (5) Wherever possible, tests shall be accomplished under actual or simulated operating conditions, including sequence of operations, for example, diesel load sequencing.	1975	51	
IEE-338-P-4	Operator Procedure Standard	6. Testing Program, page 9. 6.3 Types of Tests 6.3.1 Instrument Checks. The operability of instrument channels which have indication available shall be verified by one or more of the following: (1) Comparing readings on channels which monitor the same variable recognizing any differences in the actual process variable between sensor locations (for example, compare power channel with redundant power channels 2 and 3). (2) Comparing readings between channels which monitor the same variable and bear a known relationship to one another (for example, comparing intermediate range and source range neutron monitoring channels during a startup or shutdown when both channels indicate on scale). (3) Comparing readings between channels which monitor different variables and bear a known relationship to one another (for example, at a given power level the primary coolant outlet temperature is a certain value, or steam pressure is in a certain range).	1975	51	

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems, ANSI/IEEE Std. 338, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE-338-P-4	Operator Procedure Standard	6. Testing Program, page 9. 6.6.2 Procedure. The written procedure should contain the following: (1) The purpose of the test (2) A reference section which includes applicable mechanical or electrical drawings or both and instruction manuals with revision numbers or dates (3) A prerequisite section, including required test equipment and special communications, if required (4) A note: Once begun, a system test shall be carried through to completion and the tested system returned to service or committed to repair (5) Administrative controls (for example, obtaining permission to perform the test or informing others that the test is about to begin and its effects on the system) (6) Identification of the test input signal (for example, the nature, magnitude, and means of applying the test input signal) (7) Warnings and precautions in the procedure immediately preceding the applicable test steps (8) The anticipated response given immediately before the step which will provide the response when required as a precautionary measure. The means by which the response is to be observed shall be included in the acceptance criteria for each applicable test response (9) Clearly defined acceptance criteria (10) A requirement for notification to the responsible operator of the expected response if the test is to be performed by a person other than the operator (11) A requirement to check off or sign off procedure steps as they are performed (12) The test instrumentation to be used (for example, record the serial number and calibration due date) (13) The type of information to be given to the senior licensed operator to advise him of such things as a test termination, the results of the test, and evaluation of the results (14) Detailed instruction for removing the channel or system from service, performing the test, and restoring the channel or system to normal	1975	51	

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems, ANSI/IEEE Std. 338, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
		6. Testing Program, page 9 (continued). (15) The requirement to verify the state in which the channel or system has been left (for example, returned to service, committed for repair) (16) An explanation of test steps in complex portions of the test (17) The requirements for documentation and analysis of the test results.			

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Trail Use Standard Criteria for Post Accident Monitoring Instrumentation for Nuclear Power Generating Stations, ANSI N41.26, IEEE Std. 497, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE497-I-1	Instrumentation and Control Standard	1. Scope, page 7. This standard applies to the design of instrumentation to monitor and display required post accident conditions within the nuclear power generating station. Instrumentation addressed by the document includes that which enables the operator to: (1) identify the accident to the degree necessary for him to perform his role; (2) assess whether or not safety systems are accomplishing the required safety functions (for example, cooling the core, controlling containment pressure, etc.); (3) determine when conditions exist that require specified manual actions and monitor the results of those actions; and (4) follow the course of the accident to determine whether or not conditions are evolving within prescribed limits.	1977		
IEE497-M-1	Planning, Policy and Management Standard	4. Design Basis, page 8. A specific design basis for the post accident monitoring instrumentation shall be established for each nuclear power generation station. The design basis information thus provided shall be available, as needed, for making judgments on the adequacy of design of the post accident monitoring instrumentation. The methods for development of the specific design basis information are not within the scope of this document. The design basis shall document, as a minimum: 4.1 The generating station postulated accidents for which post accid ent monitoring instrumentation is required. 4.2 The safety systems that are required to mitigate the consequences of the postulated accidents referred to in 4.1. 4.3 The required operator actions and the conditions under which these actions are required during the post accident period. 4.4 The generating station variables to be used by the operator to: (a) identify the accidents mentioned in Section 4.1 above to the degree necessary for the operator to perform his role; (b) assess the accomplishment of the safety functions performed by the systems mentioned in Section 4.2 above; (c) guide the operator in accomplishing the required actions referred to in Section 4.3 above; and (3) follow the course of the accident to determine whether or not conditions are evolving within safe limits.	1977		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Trail Use Standard Criteria for Post Accident Monitoring Instrumentation for Nuclear Power Generating Stations, ANSI N41.26, IEEE Std. 497, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
		4. Design Basis, page 8 (continued). NOTE: Where practical, the same variable should be used for more than one of the above functions. 4.5 The portion of the post accident monitoring instrumentation that is Class IE. 4.6 The events or conditions or both which determine the time period during which the monitoring of each variable referred to in 4.4 is required. 4.7 The time after the postulated accidents when each variable referred to in Section 4.4. is first required to be monitored and the time interval during which it is required to be monitored. 4.8 The minimum number and location of the sensor(s) required for any variable referred to in Section 4.4 that have a spatial dependence. 4.9 The locations at which the information must be available to the operator and the types of information (for example: discrete state, current value of a continuous variable, long term trend) which must be presented. 4.10 The range of transient and steady-state conditions of both the energy supply and the environment (for example: voltage, frequency, electromagnetic interference, temperature, humidity, pressure, vibration, and radiation) for which provision must be incorporated to ensure adequate performance when required. 4.11 The malfunctions, accidents, or other unusual events (for example: fire, explosion, missiles, lightning, flood, earthquake, wind) which could physically damage components or could cause environmental changes leading to degradation of the performance of this instrumentation and which the design must withstand. 4.12 The maximum and minimum values and the maximum rate of change of each variable which must be accommodated by the post accident monitoring instrumentation and the maximum error within the information must be conveyed to the operator for all of the applicable conditions listed in 4.10 and 4.11 above.			

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Trail Use Standard Criteria for Post Accident Monitoring Instrumentation for Nuclear Power Generating Stations, ANSI N41.26, IEEE Std. 497, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE497-I-2	Instrumentation and Control Standard	5. General Requirements, page 8. 5.3 Display Requirements. 5.3.1 Minimizing Displays. To the extent feasible and practical, the same information display channel shall be used for normal operation and post accident monitoring.	1977		
IEE497-I-3 IEE497-O-1	Instrumentation and Control Standard Operator/System Integration Standard	5. General Requirements, page 8. 5.3.2 Location and Identification. Post accident monitoring displays shall be located accessible to the operator during the post accident period and shall be distinguishable from other displays. Post accident monitoring displays which enable the operator to determine when conditions exist that require specified manual actions, or monitoring the results of those actions, shall be located in the vicinity of the control stations used to effect the actions.	1977		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Recommended Practice for the Design of Display and Control Facilities for Central CR's of Nuclear Power Generating Stations, Std. 566, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE556-OR-1	Operator/System Integration Recommended Practice	7. Functional "Considerations", page 7. 7.1 General. The operator should be considered as one part of an integrated system that is necessary for the proper and efficient operation of a nuclear power plant.	1977		
IEE556-OR-2	Operator/System Integration Recommended Practice	7. Functional "Considerations", page 7. 7.2 Display Facilities. In support of the operator needs, the control room designer should arrange the display facilities so that the operator can readily observe the displays and analyze the status of any system.	1977		
IEE556-OR-3 IEE556-IR-1	Operator/System Integration Recommended Practice Instrumentation and Control Recommended Practice	7. Functional "Considerations", page 7. 7.2.2 Readability and Comprehension. The display equipment should provide means to facilitate operator comprehension. These include consistent use of the following: (1) Physical differentiation of data which are presented, using such techniques as color coding, size, and shape. (2) Formats keyed to and consistent with the physical representation should be used, for example, a vertical bar indicator for level. (3) Graphic displays for: flow diagrams, one-line electric diagrams, bar charts, etc.)	1977		
IEE556-OR-4	Operator/System Integration Recommended Practice	7. Functional "Considerations", page 7. 7.3.1 Control devices and their functionally associated displays should be located to facilitate operator action.	1977		
IEE556-OR-5 IEE556-IR-2	Operator/System Integration Recommended Practice Instrumentation and Control Recommended Practice	7. Functional "Considerations", page 7. 7.3.2 In determining whether control devices should be made available to the operator in the control room, the following factors should be considered: (1) the safety functions of the controlled equipment, (2) consequences of the operator not being able to take necessary action, (3) the degree of automation to be used for control, (4) the frequency of usage of the controls, and (5) the number of controls required to accomplish a given function.	1977		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Recommended Practice for the Design of Display and Control Facilities for Central CR's of Nuclear Power Generating Stations, Std. 566, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE556-OR-6	Operator/System Integration Recommended Practice	7. Functional "Considerations", page 7. 7.4 Device and Display Identification. Identification of control and display functions should be easily associated with the physical devices being monitored or controlled. Where alphanumeric identification systems are used, they should be supplementary to a functional identification.	1977		
IEE556-OR-7	Operator/System Integration Recommended Practice	7. Functional "Considerations", page 7. 7.5 Convention for Control Devices. A convention should be established to provide consistency in the operation of controls that perform similar functions, for example, control switches are to be turned clockwise to "close" (for circuit breakers).	1977		
IEE556-OR-8	Operator/System Integration Recommended Practice	7. Functional "Considerations", page 7. 7.6.2 Redundant and Diverse Information. Where a number of critical parameters require redundant or diverse displays as a means of checking the reasonability of information, the alternative information sources should be located to allow the operator to use both sources in arriving at a conclusion.	1977		
IEE556-OR-9 IEE556-CR-1	Operator/System Integration Recommended Practice Control Room Environment Recommended Practice	7. Functional "Considerations", page 7. 7.7 Area Arrangement. The normal operations area should be centrally arranged within the control room to provide the operator with surveillance and access capability to other operating areas within the control room. The emergency operations area should be readily accessible and visible from the normal operations area. This area should not be in a separate room or enclosure from the normal operations area.	1977		
IEE556-OR-10	Operator/System Integration Recommended Practice	7. Functional "Considerations", page 7. 7.8 Device Arrangement. Individual devices or groups of individual devices should be arranged to minimize operator motion including changes in direction of vision.	1977		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Recommended Practice for the Design of Display and Control Facilities for Central CR's of Nuclear Power Generating Stations, Std. 566, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE556-OR-11	Operator/System Integration Recommended Practice	7. Functional "Considerations", page 7. 7.11 Internal Security. Where display and alarm devices are provided within the central control room to alert the operator to unauthorized entry into vital areas, the devices should be clearly differentiated from any devices provided for plant functions by color, arrangement, or location.	1977		
IEE556-IR-3	Instrumentation and Control Recommended Practice	7. Functional "Considerations", page 7. 7.2.1 Accessibility. As appropriate, the operator should have information available on a "dedicated," "intermittent — periodic," or "intermittent — as called for" basis. The need for information to be displayed and its accessibility to the operator depends on: (1) the consequence of the operator not taking corrective action, (2) the importance of the data to the operator in determining the plant status, (3) the degree of automation to be used in control system design, and (4) the use of such display techniques as "display by exception."	1977		
IEE556-IR-4	Instrumentation and Control Recommended Practice	7. Functional "Considerations", page 7. 7.2.3 Abnormal Conditions. The operator should be alerted to abnormal or unsafe conditions or significant changes in the plant and its process systems or safety systems or both.	1977		
IEE556-IR-5	Instrumentation and Control Recommended Practice	7. Functional "Considerations", page 7. 7.2.3.2 System Modes. Alarms should also be terminated or suppressed during modes of operation when they would be meaningless, due to changes in the operating mode (such as startup, power operation, shutdown, etc.), so that information priority for the current mode of operation can be readily assessed.	1977		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Recommended Practice for the Design of Display and Control Facilities for Central CR's of Nuclear Power Generating Stations, Std. 566, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE556-IR-6	Instrumentation and Control Recommended Practice	7. Functional "Considerations", page 7. 7.3.3 Where the controls of equipment or devices which are part of safety systems can be transferred to points of control outside the control room, the mode of the active control should be indicated in the control room.	1977		
IEE556-IR-7	Instrumentation and Control Recommended Practice	7. Functional "Considerations", page 7. 7.6 Display and Control Facilities — Special. Special requirements such as safety surveillance, post accident monitoring, and remote shutdown should be considered in usage analysis described in Section 6.	1977		
IEE556-IR-8	Instrumentation and Control Recommended Practice	7. Functional "Considerations", page 7. 7.6.1 Safety System Status. The operator should be clearly informed of the status of the safety system by means of a display. This display should be used to enhance the normal plant administrative procedures.	1977		
IEE556-PR-1	Operator Procedure Recommended Practice	7. Functional "Considerations", page 7. 7.2.3.3 Limit Monitoring. In addition to normal equipment protective limits, plant operational limits established by technical specifications and by plant administrative procedures shall be monitored by the operator.	1977		
IEE556-MR-1	Policy, Planning and Management Recommended Practice	7. Functional "Considerations", page 7. 7.9 Equipment or System Status. Consideration should be given to provide indication when non-safety-related equipment is taken out of service for maintenance, calibration, or inspection, and when it is returned to service.	1977		
IEE556-ER-1	Operator Support Equipment Recommended Practice	7. Functional "Considerations", page 7. 7.10 Communications. The methods provided for communication between the operator and various other personnel should not divert the operator from his principal duties.	1977		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Overpressure Protection of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary, ANSI/ANS 56.3, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS56.3-I-1	Instrumentation and Control Standard	4. Instrumentation and Controls, page 3. 4.2 Design Criteria 4.2.1 Standards Documents. The instrumentation and controls for any particular overpressure protection system under consideration shall be designed in accordance with the applicable IEEE Standards consistent with safety classification (3-10)[4] 4.2.2 Additional Criteria. The following criteria supplement those standards referenced in 4.2.1. (1) Power operated valves shall be capable of either remote operation from the Control Room or local operation, both subject to intervention by appropriate interlocks. Power operated valves with local control only shall be treated as manual valves and locked closed. (2) Power operated valves shall be provided with automatic remote position (open/closed) indication in the Control Room. Information regarding the position (open/closed) of manual valves shall also be displayed in the Control Room. (3) Control Room indication shall be provided to indicate when isolation is necessary. (4) The process variables to be sensed may include, but not be limited to the following: (a) High pressure system pressure with the associated set point to prevent opening of the isolation valves (b) High pressure system pressure with associated set point to initiate automatic isolation, alarm or both (c) Low pressure system pressure with associated set point to initiate automatic isolation, alarm or both.	1974	52	

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Pressurized Water Reactor Containment Ventilation Systems ANSI/ANS 56.6, 1978.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS56.6-I-1	Instrumentation and Control Standard	3. Containment Air Cooling System (CACS), page 4. 3.4.3.5 Instrumentation and Control. The CACS fans and applicable control devices shall be operable from the control room. Fan operating status indication shall be provided in the control room and an alarm shall sound in the control room if a running fan stops.	1977	53	
ANS56.6-I-2	Instrumentation and Control Standard	4. Purge Supply and Exhaust Systems, page 7. 4.4.3.5 Instrumentation and Control. Containment isolation valves and system fans shall be capable of remote manual operation from the control room. Their operational status shall be displayed in the control room. Containment isolation signals or high radiation levels shall close the PSES containment isolation valves and should stop the fans automatically. Differential pressure instruments shall be provided to indicate changes in air pressure drop across each filter bank unit in the main assembly.	1977	53	
ANS56.6-I-3	Instrumentation and Control Standard	4. Purge Supply and Exhaust Systems, page 7. Instrumentation required to isolate the PSES upon a high radiation signal due to a refueling fuel handling accident shall be redundant, satisfy the single failure criteria, and be SSE qualified. The monitor to detect this isolation function should be fast acting relative to the monitor location, exhaust duct velocity and PSES isolation valve closure time.	1978		
ANS56.6-I-4	Instrumentation and Control Standard	6. Reactor Cavity Cooling System, page 12. 6.4.3.5 Instrumentation and Control. The RCCS fans shall be operable from the control room. Fan running lights shall be provided in the control room and an alarm shall sound in the control room if the running fan should stop. Temperature sensing devices should be provided at appropriate locations to provide an alarm in the control room if temperatures approach the design maximum value.	1977	53	

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: ____________________

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS56.6-I-5	Instrumentation and Control Standard	7. Containment Heating System, page 13. 7.4.3.5 Instrumentation and Control. The CHS fans should be controlled by thermostats located in their respective areas. Switches should be provided to enable the fan to be controlled locally.	1978		
ANS56.6-I-6	Instrumentation and Control Standard	8. Containment Cleanup System, page 14. 8.4.3.5 Instrumentation and Control. Instrumentation shall be furnished to indicate changes in air pressure drop across each filter bank.	1978		
ANS56.6-I-7	Instrumentation and Control Standard	9. Containment Compartment Cooling Systems, page 15. 9.4.3.5 Instrumentation and Control. The CCCS fans should be controlled from the control room. Fan running lights should be provided in the control room and an alarm should sound in the control room if any running fan should stop. Switches should be provided to enable the fan to be started and stopped at a local station.	1978		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Proposed American National Standard Criteria for Safety-Related Operator Actions, ANSI N660/ANS-51.4, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.4-O-1	Operator/System Integration Standard	3.0 General Requirements for Operator Actions, page 6. 3.1 Safety system response to design basis events shall be initiated by automatic protection systems if the protective action must be initiated earlier than allowed by the Time Test I intervals given in Section 5.	1977		
ANS51.4-O-2	Operator/System Integration Standard	3.0 General Requirements for Operator Actions, page 7. 3.2 Safety system response to design basis events may be initiated by required operator action(s) if all of the requirements of this document are met, particularly the time test requirements of section 5 herein.	1977		
ANS51.4-O-3	Operator/System Integration Standard	3.0 General Requirements for Operator Actions, page 7. 3.3 After automatic or operator initiation of the safety systems, required operator actions may be used for initiation of subsequent protective actions required in the sequence of the design basis events if all the requirements of this document are met.	1977		
ANS51.4-O-4 ANS51.4-I-1	Operator/System Integration Standard Instrumentation and Control Standard	3.0 General Requirements for Operator Actions, page 7. 3.4 Required operator actions or sequences of actions shall only be used where there is time and information available for the operator to recognize an error and where equipment and process design permits corrective action.	1977		
ANS51.4-4-O-5	Operator/System Integration Standard	3.0 General Requirements for Operator Actions, page 7. 3.5 The number of required operator actions or sequences of actions shall be minimized to the extent that the operators have sufficient time to monitor the plant status, and perform optional operator actions.	1977		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: ______________________________

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.4-P-1	Operator Procedure Standard	3.0 General Requirements for Operator Actions, page 7. 3.5 The number of required operator actions or sequences of actions shall be minimized to the extent that the operators have sufficient time to monitor the plant status, and perform optional operator actions.	1977		
ANS51.4-O-6	Operator/System Integration Standard	3.0 General Requirements for Operator Actions, page 7. 3.6 Protective actions that require frequent or continuous monitoring or adjustment shall be automated where practical.	1977		
ANS51.4-O-7	Operator/System Integration Standard	3.0 General Requirements for Operator Actions, page 7. 3.7 The number of the required operator actions specified at any point in time shall be limited to a value that can be conducted by the number of operators available.	1977		
ANA51.4-I-2	Instrumentation and Control Standard	4.0 Locations for Operator Actions and Operator Environmental Protection, page 7. 4.1 All operator actions required in less than 30 minutes following design basis events shall be capable of being performed from the control room.	1977		
ANS51.4-P-2	Operator Procedure Standard	4.0 Locations for Operator Actions and Operator Environmental Protection, page 7. 4.1 All operator actions required in less than 30 minutes following design basis events shall be capable of being performed from the control room.	1977		
ANS51.4-I-3 ANS51.4-P-3 ANS51.4-M-1	Instrumentation and Control Standard Operator Procedure Standard Planning, Policy and Management Standard	4.0 Locations for Operator Actions and Operator Environmental Protection, page 8. 4.5 It shall be a design objective to (a) minimize the number of required operator actions that must be performed from locations outside of the control room, and (b) minimize the number of locations outside the control room at which required operator actions are performed.	1977		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: ______________________________

____

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.4-I-4	Instrumentation and Control Standard	7.0 Information Availability, page 14. 7.1 The operator shall be provided with clearly presented readout information, at the required time for him to assess the need for a particular protective action without significant diagnoses.	1977		
ANS51.4-I-5	Instrumentation and Control Standard	7.0 Information Availability, page 14. 7.2 Each channel of readout information that indicates the initiation (at t_e) of a design basis event shall include both an indicator and an audible alarm, such as an annunciator. More than one variable may be required to identify the initiation of a design basis event.	1977		
ANS51.4-I-6	Instrumentation and Control Standard	7.0 Information Availability, page 15. 7.3 Each channel of readout information that indicates the need (at t_a) for a required operator action that must be initiated within 30 minutes after the operator action alarm (i.e., (t_i - t_a) F 30 minutes) shall include both an indicator and an audible alarm, such as an annunciator.	1977		
ANS51.4-I-7 ANS51.4-P-4	Instrumentation and Control Standard Operator Procedure Standard	7.0 Information Availability, page 15. 7.4 Each channel of readout information that indicates the need for a required operator action that need not be initiated until 30 minutes or more after the operator action alarm (i.e., (t_i - t_a) f 30 minutes) shall include either an indicator and an audible alarm, or an indicator supplemented by an emergency procedure. This procedure shall include an estimate of the time at which each required operator action must be initiated.	1977		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: ______________________________

____

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.4-I-8	Instrumentation and Control Standard	7.0 Information Availability, page 15. 7.5 Readout information shall be provided which indicates that each action controlled by an operator manipulation has been correctly initiated.	1977		
ANS51.4-I-9	Instrumentation and Control Standard	8.0 Reliability of Instrumentation and Controls, page 15. 8.2 A minimum of three channels of readout information shall be provided to indicate the need for required operator actions that affect more than one train of safety system equipment.[9] This requirement can be reduced to two channels if the operator can always take a safe action when faced with a disagreement in display information or if appropriately qualified indications of diverse related variables are available to give similar information.	1977		
ANS51.4-I-10	Instrumentation and Control Standard	8.0 Reliability of Instrumentation and Controls, page 16. 8.3 Where at least two trains of safety system equipment are provided, a minimum of one channel of readout information per train shall be provided to indicate the need for required operator actions that would only affect one train of the safety system equipment.	1977		
ANS51.4-M-2	Planning, Policy and Management Standard	9.0 Safety Analyses and Emergency Procedures, page 16. 9.1 The time delays, time margins, required operator actions, and their associated instrumentation, controls, and locations (if outside the main control room) shall be documented in the safety analysis for each design basis event. 9.2 No credit shall be taken in the safety analysis of design basis events for optional or unplanned operator actions.	1977		
ANS51.4-P-5	Operator Procedure Standard	9.0 Safety Analyses and Emergency Procedures, page 16. 9.3 Required operator actions shall be included in the formal plant emergency procedures.[10] The discrete manipulations (from Time Test 2) shall be identified in the procedures.	1977		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Single Failure Criteria for PWR Fluid Systems
ANS 51.7, 1976.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS51.7-I-1	Instrumentation and Control Standard	3. Rules for Application of the Single Failure Criteria 3.9 The unit design shall be such that active components of safety systems and their related service systems can be proved operational by scheduled periodic operational tests and by automatic or manual operational status indications.	1976		
ANS51.7-M-1	Policy, Planning and Management Standard	3. Rules for Application of the Single Failure Criteria 3.10 The designer shall consider in his design operator error as a potential single failure in addition to the initiating event. If suitable time and means for detection and diagnosis of operator error are provided, correction of the error may be assumed.[7]	1976		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Single Failure Criteria for Light Water Reactor (LWR) Safety Related Fluid Systems, ANSI/ANS-58.9, Draft 4, 1979.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS58.9-M-1 ANS58.9-O-1	Planning, Policy and Management Standard Operators/System Integration Standard	3. Rules for Application of the Single Failure Criteria, page 6. 3.10 The designer shall consider in his design an operator error as a potential single active failure in addition to the initiating event.	1979		
ANS58.9-I-1 ANS58.9-O-2 ANS58.9-M-2 ANS58.9-P-1	Instrumentation and Control Standard Operator/System Integration Standard Planning, Policy and Management Standard Operator Procedure Standard	3. Rules for Application of the Single Failure Criteria, page 6. 3.11 If suitable time and means for detection, diagnosis, and correction of single failures are provided, operator actions for mitigation of consequences of the single failure shall be allowed.	1979		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Emergency Control Centers for Nuclear Power Plants

ANSI/ANS 3.7.3, 1979.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS3.7.2-I-1 ANS3.7.2-E-1	Instrumentation and Control Standard Operator Support Equipment Standard	3. Types of Emergency Control Centers, page 1. 3.1.2 Communications. The nuclear plant control room shall have redundant two-way communications with the emergency control center, company headquarters, and with appropriate off-site support agencies responsible for initial actions. At a minimum, the communications with the various emergency control centers shall include normal telephone communications and an alternate means. The alternate method may include, depending on the distances involved, sound-powered telephones, two-way radios, microwave, or the national warning system (NAWAS).	1978	54	
ANS3.7.2-I-2	Instrumentation and Control Standard	3. Types of Emergency Control Centers, page 1. 3.1.3 Instrumentation and Equipment.[3] The instrumentation and equipment requirements for the control room shall include but not be limited to (1) instrumentation to evaluate the principal plant variables indicative of the plant status and future conditions, (2) instrumentation to evaluate the release rate of radionuclides and the meteorological conditions (i.e., wind speed, wind directions, and stability) at the site, (3) access to instrumentation for radiological surveillance, and (4) equipment necessary to ensure the habitability of the nuclear plant control room during the course of an accident.	1978	54	
ANS3.7.2-P-1	Operator Procedure Standard	3. Types of Emergency Control Centers, page 1. 3.1.4 Decisional Aids. The emergency personnel shall have access to prepared isopleth dose curves (or their equivalent) for a broad range of representative release rates or source terms and meteorological conditions. Given a monitored or calculated source term and the meteorological conditions, the information from these curves can assist in providing an early estimate of the projected on- and off-site radiological impact and the time available to implement protective actions.	1978	54	

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Earthquake Instrumentation Criteria for Nuclear Power Plants, ANSI/ANS 2.2, 1978.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
ANS2.2-I-1	Instrumentation and Control Standard	5. Instrument Characteristics, page 4. 5.5.6 Miscellaneous. The time-history accelerograph on the containment foundation and the containment structure shall be interconnected for common starting and common timing, and shall contain provision for external alarm to indicate actuation.	1977	55	
ANS2.2-I-2	Instrumentation and Control Standard	6. Instrumentation Station Installation, page 5. 6.5 Remote Indication. Upon actuation of any time-history accelerograph, seismic switch or response spectrum switch, a remote indication, preferably in the control room, shall be activated. The remote indication for the seismic switch required in 4.1.4 and the response spectrum switch required in 4.1.5, however, shall be annunciator(s) in the control room.	1977	55	
ANS2.2-I-3	Instrumentation and Control Standard	6. Instrumentation Station Installation, page 5. 6.6 Instrumentation Station Accuracy. Instruments and their interconnections shall be installed so that the instrumentation station shall be capable of providing data with an overall error of not more than $\pm 5\%$ at full scale, changing linearly to $\pm 1.5\%$ of full scale at 0.01g, over the appropriate range of environmental conditions, such as temperature, humidity, pressure, vibration and radiation.	1977	55	

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations, IEEE Std. 323, 1974.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE-323-I-I	Instrumentation and Control Standard	6. Qualification Procedures and Methods, page 10. 6.2 Equipment Performance Specifications. Electric equipment specifications shall define the equipment's Class IE requirements and shall include as applicable: (6) Control, indicating, and other auxiliary devices contained in the equipment or external to the equipment and required for proper operation.	1974		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: IEEE Trial-Use Standard-Criteria for Safety Systems for Nuclear Power Generating Stations, Std. 603, 1977.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE-603-O-1 IEE-603-I-1	Operator/System Integration Standard Instrumentation and Control Standard	4. Safety System Functional and Design Requirements, page 13. 4.2.1 Means shall be provided in the control room to implement manual initiation at the system level of the automatically initiated protective actions. The means provided shall minimize the number of discrete operator manipulations and shall depend on the operation of a minimum of equipment.	1968	56	
IEE-603-O-2	Operator/System Integration Standard	4. Safety System Functional and Design Requirements, page 13. 4.11.4 Location. Information displays shall be located accessible to the operator. Information displays provided for manually initiated protective actions shall be visible from the location of the controls used to effect the actions.	1977		
IEE-603-P-1 IEE-603-I-2	Operator Procedure Standard Instrumentation and Control Standard	4. Safety System Functional and Design Requirements, page 13. 4.2.3 Means shall be provided to implement the manual actions necessary to maintain safe conditions after the proective actions are completed as specified in 3.10. The number of available qualified operators, the information provided to these operators, the actions required of these operators, and the quantity and location of associated displays and controls shall be appropriate for the time period within which the actions must be accomplished. Such displays and controls shall be located in areas that are accessible and in an environment suitable for the operator.	1977		
IEE-603-P-2	Operator Procedure Standard	4. Safety System Functional and Design Requirements, page 13. 4.4 Completion of Protective Action. The safety system shall be designed so that, once initiated automatically or manually, the intended sequence of protective actions at the system level shall continue until completion. Deliberate operator action shall be required to return the safety system to normal. This requirement shall not preclude the use of equipment protective devices or the provision for those deliberate operator interventions which are identified in 3.10 of the design basis.	1968	56	

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: ______________________

____

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE-603-I-3	Instrumentation and Control Standard	4. Safety System Functional and Design Requirements, page 13. 4.2.2 Means shall be provided in the control room to implement manual initiation of the protective actions identified in 3.5 that have not been selected for automatic initiation under 4.1.	1968	56	
IEE-603-I-4	Instrumentation and Control Standard	4. Safety System Functional and Design Requirements, page 13. 4.11 Information Displays 4.11.1 Displays for Protective Actions Initiated Solely by Manual Means. The display instrumentation provided for the manually initiated actions required for the safety system to accomplish its protective function shall be part of the safety system. The design shall minimize the possibility of anomalous indications which could be confusing to the operator.	1968	56	
IEE-603-I-5	Instrumentation and Control Standard	4. Safety System Functional and Design Requirements, page 13. 4.11.2 System Status Indication. The display instrumentation provided for safety system status indication need not be part of the safety system. The display instrumentation shall provide accurate, complete, and timely information pertinent to safety system status. This information shall include indication and identification of protective actions at the channel level and the system level. The design shall minimize the possibility of anomalous indications which could be confusing to the operator.	1968	56	
IEE-603-I-6	Instrumentation and Control Standard	4. Safety System Functional and Design Requirements, page 13. 4.11.3 Indication of Bypasses. If the protective actions of some part of the safety system have been bypassed or deliberately rendered inoperative for any purpose, continuing indication of this fact at the system level shall be provided in the control room.	1968	56	
IEE-603-O-3	Operator/System Integration Standard	6. Protective Action System Functional and Design Requirements, page 17. 6.1 Manual Initiation. If manual initiation of any actuated component in the protective action system is required to fulfill a design basis objective, the additional design features in the protective action system necessary to accomplish such manual initiation shall not defeat the requirements of 4.2 or 4.3.	1977		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

HUMAN ENGINEERING AND RELATED STANDARDS AND RECOMMENDED PRACTICES

Reference: Proposed IEEE Criteria for Nuclear Power Plant Protection System, Std. 279, 1968.

Number	Type of Standards or Recommended Practices	Language of Standards or Recommended Practices	Earliest Known Publication Date (1)	Other Reference (2)	Notes (3)
IEE279-I-1 IEE279-O-1	Instrumentation and Control Standard Operator/System Integration Standard	4. Requirements, page 4. 4.9 Capability for Sensor Checks. Means shall be provided for checking, with a high degree of confidence, the operational availability of each system input sensor during reactor operation. (a) by perturbing the monitored variable; or (b) within the contraints of paragraph 4.11, by introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable; or (c) by cross checking between channels that bear a known relationship to each other and that have read-outs available.	1968		
IEE279-O-2	Operator/System Integration Standard	4. Requirements, page 4. 4.12 Operating Bypasses. Where operating requirements necessitate automatic or manual bypass of a protective function, the design shall be such that the bypass will be removed automatically whenever permission conditions are not met. Devices used to achieve automatic removal of the bypass of a protective function are part of the protection system and must be designed in accordance with these Criteria.	1968		

NOTES: (1) 1967 or more recent.
(2) If checked, see list of references attached.
(3) If checked, see list of notes attached.

APPENDIX O

HUMAN ENGINEERING ASPECTS OF CONTROL ROOM DESIGN

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Controls

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Selection of Controls	Control selection and design is dependent on: o Distribution of load, such that operators limbs are not overburdened o Control capabilities are paired to functional requirements: - continuous variables - discrete variables - precision requirements - system activation - data entry - quantitative setting	MIL-STD-1472B Van Cott and Kinkade McCormick Chapanis AFSC DH 1-3	 Yes Yes	
Direction of Control Movement	Following are considerations relevant to control direction and movement: o Consistency with direction of movement of associated displays o Direction of movement consistent with orientation of the operator	MIL-STD-1472B Van Cott and Kinkade Chapanis AFSC DH 1-3 MSFC-STD-512	 Yes Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Controls

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Arrangement and Grouping	Considerations related to arrangement and grouping are as follows:	MIL-STD-1472B Van Cott and Kinkade McCormick Chapanis		
	o Controls grouped according to sequential relations in operation		Yes	
	o Primary controls located in most favorable position with respect to ease of reaching and operating		Yes	
	o Recurring control groups similar in layout from panel to panel		Yes	
	o Minimum/maximum control spacing addressed as part of design		Yes	
Coding	o Selection of coding methods (shape, size, color) consistent with coding requirements and other factors (ambient light, etc.)	MIL-STD-1472B Van Cott and Kinkade Bioastronautics Data Book Chapanis	Yes	
	o Coding modes (size, shape, color) consistent with system - functionally similar controls have same coding		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Controls

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Color Coding	Color coding of controls is used only when required. Otherwise, controls are black on gray.	MIL-STD-1472B Van Cott and Kinkade Chapanis MSFC-STD-512	Yes	
Control Compatibility with Hardware	Controls should be compatible with any hardware used	MIL-STD-1472B Malone MSFC-STD-512		
Prevention of Accidental Activation	Considerations are as follows:	MIL-STD-1472B Van Cott and Kinkade Malone Chapanis		
	- location of controls		Yes	
	- design of controls (guards, spring loading, etc.)		Yes Yes	
	- controls designed to prevent accidental activation should still be operable		Yes	
General Control Design Considerations	Following are control design features which should be considered during control design/selection:	MIL-STD-1472B Van Cott and Kinkade Chapanis McCormick Malone		
	- minimum/maximum number of switch positions		Yes	
	- presence of detents		Yes	
	- switch resistance		Yes	
	- switch labels		Yes	
	- switch legends		Yes	
	- label/legend contrast		Yes	
	- label parallax		Yes	
	- control dimensions		Yes	
	- control resistance		Yes	
	- control displacement		Yes	
	- control separation		Yes	
	- guards/barriers		Yes	
	- control size/shape		Yes	
	- control location		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Controls

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
General Control Design Considerations (continued)	- control illumination		Yes	
	- control luminance		Yes	
	- visibility		Yes	
	- associated displays		Yes	
	- direction of control movement		Yes	
	- feedback		Yes	
	- orientation to the operator		Yes	
	- coding, size/shape/color/position		Yes	
	- sensitivity		Yes	
	- speed of response		Yes	
	- reliability		Yes	
	- stability		Yes	
	- accuracy		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Visual Displays

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Information Displayed	o Provides the operator with clear indications of system conditions which prompts operator actions, decisions	MIL-STD-1472B Van Cott and Kinkade AFSC DH 1-3 Chapanis McCormick	Yes	
	o Specific areas to be addressed include: - content, in terms of what is to be displayed - precision required in the information displayed - information format		Yes	
	o Displayed information should not be redundantly displayed unless required at different operating stations		Yes	
	o Display failure should: - be immediately apparent to the operator - not cause a failure in the operability of the equipment associated with the display		Yes	
	o Does not exceed operator capacity		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Visual Displays

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Location and Arrangement	Displays should be located such that:	MIL-STD-1472B Van Cott and Kinkade Malone McCormick		
	o They may be read in the normal operating position		Yes	
	o They require no tools to read (such as ladders, flashlights, etc.)		Yes	
	o They are oriented to the line of sight of the operator in the normal operator position		Yes	
	o Display surfaces do not reflect ambient light		Yes	
	o They are grouped according to: - usage rates - operational sequence - importance		Yes	
	o Viewing distance is accounted for in the design		Yes	
Coding	Coding should be used to facilitate:	MIL STD-1472B Van Cott and Kinkade Chapanis AFSC DH 1-3 Malone MSFC-STD-512		
	- display discrimination		Yes	
	- identification of functionally similar displays		Yes	
	- identification of display relationships		Yes	
	- identification of critical information within a display		Yes	
	- information processing		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Visual Displays

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Display Use	o Display type selection (use) depends on the characteristics of the information to be displayed: - continues control - status monitoring - briefing/alerting - search/identification - decision making - trend analysis	MIL-STD-1472B Van Cott and Kinkade Chapanis	Yes	
General Display Characteristics to be Considered as part of CR design	o Indicator lights should not be used (in the extinguished mode) to indicate a system "go" condition	MIL-STD-1472B Van Cott and Kinkade Bioastraunautics Data Book	Yes	
	o These considerations include:		Yes	
	- information displayed		Yes	
	- functional grouping		Yes	
	- luminance		Yes	
	- luminance control		Yes	
	- display operability testing		Yes	
	- contrast between legends and background		Yes	
	- color coding		Yes	
	- parallax		Yes	
	- multiple legends		Yes	
	- visbility		Yes	
	- visual environment		Yes	
	- signal rate		Yes	
	- resolution		Yes	
	- discriminability		Yes	
	- legends		Yes	
	- character sizes		Yes	
	- symbology		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Visual Displays

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Color Coding	Conveying information by associating color with system information: - red — not within tolerance conditions - flashing red — emergency condition - yellow — marginal condition - green — positive indication of system operability - white — alternate functions active	MIL-STD-1472B Van,Cott and Kinkade AFSC DH 1-3 Chapanis McCormick Malone MSFC-STD-512	Yes	
Display Characteristics to be Addressed as Part of Design	o Transilluminated displays - legends - backlighting - intensity controls - lamp redundancy - lettering . font . character sizes - color coding - flash rates (as applicable) - visibility - legibility - symbology - size/shape	MIL-STD-1472B Chapanis	Yes	
	o Legend lights - color - labels/font/sizes - spacing - size shape		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Visual Displays

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Display Characteristics (continued)	o Scale Indicators - moving pointer - moving scale - accuracy - parallax - labels, legends - tolerance markings - graduation - numerical size - start/end points - size/shape/location - numeric progression - scale break (guages) - nominal (when equipment functioning properly) pointer position - viewing distance		Yes	
	o CRTs - viewing distance - screen luminance - ambient illumination - reflected glare - symbology - edit/input devices		Yes	
	o LEDs - applications - readability - colors/color coding - intensity controls - test provisions		Yes	
	o Counters, plotters, flags - snap action vs. continuous movement - rate of movement - direction of movement - resets - parallax - color - illumination - contrast - visibility		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Visual Displays

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Display Characteristics (continued)	- mounting - test provisions - size/shape			
	o Printers - form of information presentation - take-up provisions - annotations - visibility - illumination - contrast		Yes	
Display Errors	Display design should address the following error types:	McCormick Malone Chapanis		
	- temporal		Yes	
	- selection (wrong display read)		Yes	
	- interpretation		Yes	
	- reading		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Audio Displays

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Applications of Audio Displays	Audio displays considered for use under the following conditions: - information presented is transitory requiring immediate or time based operator response - visual channels are overburdened or otherwise unavailable for required operator attention - required redundnacy to visual indications	MIL-STD-1472B Van Cott and Kinkade Chapanis	Yes	
Warning Signal Characteristics	Design considerations are as follows:	MIL-STD-1472B Van Cott and Kinkade		
	- tonal frequency		Yes	
	- intensity		Yes	
	- alerting capability		Yes	
	- ambient noise		Yes	
	- discriminability		Yes	
	- volume control		Yes	
	- provision to shut off alarms		Yes	
	- test provisions		Yes	
	- duration of signals		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: General HFE in Systems

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Standardization	Uniformity within systems and subsystems with similar functions	MIL-STD-1472B Van Cott and Kinkade Chapanis	Yes	
Function Allocation	Allocation of system functions to men and/or machines based on relative:	MIL-STD-1472B Van Cott and Kinkade Chapanis McCormick		
	o Precision/sensitivity		Yes	
	o Time		Yes	
	o Safety		Yes	
	o Skill requirements/ capabilities		Yes	
	o Cost		Yes	
	o Performance/effectiveness		Yes	
	o Human/machine reliability		Yes	
Human Engineering Design	Designing to enhance human performance through (where possible):	MIL-STD-1472B Van Cott and Kinkade		
	o Controlling atmospherics		Yes	
	o Controlling noise, shock, etc.		Yes	
	o Environmental protection		Yes	
	o Providing adequate operator space		Yes	
	o Design of communication networks		Yes	
	o Workspace layout		Yes	
	o Workspace illuminated		Yes	
	o Design of life support equipment		Yes	
	o Design of emergency systems		Yes	
	o Design of information processing and decision systems		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: General HFE in Systems

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Fail Safe Design	Providing a fail safe design for those areas where human error and/or equipment malfunctions may have catastrophic effects on system operability	MIL-STD-1472B	Yes	
Simplicity of Design	Providing as simple an operational design as possible, consistent with system functional requirements	MIL-STD-1472B	Yes	
Safety Design	Minimizing potential of human error during system operation and maintenance	MIL-STD-1472B Chapanis Van Cott and Kinkade	Yes	
User Acceptance	Enhancing user confidence and acceptance	AFSC DH 1-3	Yes	
Training Requirement Reduction	Training requirements reduced through simplicity of design	AFSC DH 1-3	Yes	
Operator Performance	Minimizing human error along the dimensions of:	Malone Van Cott and Kinkade		
	o Time		Yes	
	o Motor responses		Yes	
	o Decisions		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Control/Display (C/D) Integration

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
C/D Compatibility	Controls and displays functionally compatible, and minimize mental involvement on the part of the operator	Van Cott and Kinkade MIL-STD-1472B Chapanis AFSC DH 1-3 MSFC-STD-512 McCormick	Yes	
C/D Relationship	Physical proximity of functionally related controls and displays	MIL-STD-1472B McCormick Chapanis	Yes	
C/D Design	C/D integration through functional grouping, similarity of grouping for recurrent panels, C/D coding, C/D labeling, framing, etc.	MIL-STD-1472B AFSC DH 1-3 Chapanis McCormick Van Cott and Kinkade	Yes	
C/D Precision	Control precision consistent with system requirements, display precision consistent with associated control precision	MIL-STD-1472B McCormick Van Cott and Kinkade Chapanis	Yes	
Feedback	Postive indication of system response to control activation	MIL-STD-1472B Van Cott and Kinkade McCornick Chapanis	Yes	
C/D Functional Group Arrangements	o Controls and displays positioned according to: - sequence of use (left to right or top to bottom positioning) - frequency of use - importance	MIL-STD-1472B Chapanis Van Cott and Kinkade McCormick AFSC DH 1-3 Malone	Yes	
	o Recurring groups of C/Ds are consistent in arrangement		Yes	
	o Controls positioned under associated displays		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Control/Display (C/D) Integration

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
C/D Movement Relationships	Control and display movement relationships are consistent in terms of: - direction of movement - direction to increase/ decrease, cycle, on/off etc.	MIL-STD-1472B Van Cott and Kinkade McCormick Chapanis	Yes	
C/D Ratios	Ratios of C/D excursions consistent with functional requirements while minimizing time required to make and verify desired control movement	MIL-STD-1472B Van Cott and Kinkade McCormick Chapanis	Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Data Entry Devices

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Data Entry Devices	o Use of keyboards, etc., used to make data entries to system control systems and processors, analyzers, and so on	Van Cott and Kinkade MSFC-STD-512	Yes	
	o Specific areas to be addressed include:			
	- clarity (output)		Yes	
	- readability (output)		Yes	
	- format requirements (input)		Yes	
	- data type requirements . numeric . alphanumeric		Yes	
	- input/output redundancy		Yes	
	- feedback		Yes	
	- data uses (output)		Yes	
	- data manipulation requirements		Yes	
	- encoding		Yes	
	- data entry devices keyboards, etc.)		Yes Yes	
	- data output devices		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Labeling

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
General	Controls, displays, and other components that must be located, read, etc., should be labeled such that rapid and accurate performance is permitted	MIL-STD-1472B McCormick MSFC-STD-512 Chapanis	Yes	
Label Characteristics	Label characteristics extend to:	MIL-STD-1472B MSFC-STD-512 AFSC DH 1-3 Malone Van Cott and Kinkade		
	- accuracy required		Yes	
	- time required to read labels		Yes	
	- distance at which labels should be read		Yes	
	- ambient illumination levels		Yes	
	- label criticality		Yes	
	- consistency with other labels		Yes	
Orientation and Location	o Labels should be horizontally oriented	MIL-STD-1472B Van Cott and Kinkade Malone AFSC DH 1-3	Yes	
	o Labels placed near item identified/described		Yes	
	o Labels consistent, standardized		Yes	
Contents	o Labels describe functions of equipment items	MIL-STD-1472B Van Cott and Kinkade Malone	Yes	
	o Properly abbreviated		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Labeling

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Qualities	Following are characteristics relevant to label qualities:	MIL-STD-1472B Van Cott and Kinkade Malone MSFC-STD-512		
	- brevity		Yes	
	- familiarity		Yes	
	- visibility		Yes	
	- legibility . contrast . character style . application(decal, etc.) . reflection		Yes	
	- visual access (extent obscured)		Yes	
	- label background		Yes	
Design of Label Characters	Design of label characters entails addressing:	MIL-STD-1472B Malone MSFC-STD-512		
	- character color		Yes	
	- requirements for dark adaptation		Yes	
	- style/font		Yes	
	- letter width		Yes	
	- letter height		Yes	
	- stroke width		Yes	
	- character spacing		Yes	
	- word spacing		Yes	
	- line spacing		Yes	
	- label size vs. luminance		Yes	
	- label size vs. viewing distance		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Labeling

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Equipment Labeling	o Assemblings, components and parts labeled, clearly, by name or symbol o Additional factors include: - location of labels - terms used	MIL-STD-1472B MSFC-STD-512 Malone Van Cott and Kinkade McCormick Chapanis	Yes Yes	
Labeling of Controls and Displays	Display labeling charactersitics to be addressed include: - simplicity - similarity both in terms of: . redundant controls . similar controlling functions (on/off) - control/display relationships - location of labels - label size graduations	MIL-STD-1472B MSFC-STD-512 Malone Van Cott and Kinkade McCormick Chapanis	Yes Yes Yes Yes Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Workspace

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Workspace	Includes aspects of the physical environment from which control (partial or whole) of the system is affected. Encompasses:	MIL-STD-1472B NASA Van Cott and Kinkade MSFC-STD-512 Chapanis McCormick Malone	Yes	
	- control/display placements		Yes	
	- panel locations		Yes	
	- console dimensions and configurations		Yes	
	- stairs, ramps, etc.		Yes	
	- ingress, egress		Yes	
	- visual envelopes		Yes	
	- procedural efficiency		Yes	
	- shared operations		Yes	
	- workspace traffic		Yes	
	- environmental factors such as temperature, humidity		Yes	
	- workspace safety		Yes	
Standing Operations	Considerations for standing operations include:	MIL-STD-1472B Van Cott and Kinkade McCormick Malone		
	- work surface		Yes	
	- control and display placement		Yes	
	- mobility requirements and:		Yes	
	. depth of work area		Yes	
	. lateral work space		Yes	
	. workspace layout		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Workspace

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Seating	Considerations include: - seating capability with human anthropometry - seat adjustments - backrests, armrests - cushioning - knee room - control/display placement	MIL-STD-1472B Van Cott and Kinkade Malone	Yes	
Console Design	Console design as related to workspace involves the consideration of:	MIL-STD-1472B NASA Van Cott and Kinkade		
	- visibility requirements	MSFC-STD-512	Yes	
	- mobility requirements	Chapanis	Yes	
	- panel space requirements	McCormick Malone	Yes	
	- console volume		Yes	
	- panel/console: . width . angles . height . viewing angles . shelf heights . writing surfaces		Yes	
	- task networks/procedures		Yes	
	- population stereotypes		Yes	
Stairs, Ladders, and Ramps	Design areas requiring consideration include: - handrails - guardrails - provisions for hand carrying of equipment - ramp cleating - traffic (personnel and vehicle) - platforms	MIL-STD-1472B Malone McCormick	No	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Workspace

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Ingress/Egress	Considerations include:	MIL-STD-1472B		
	- sliding door design	NASA	No	
	- hatches	MSFC-STD-512	No	
	- force to open	Malone	No	
	- configurations		No	
	- dimensions		No	
Environment	Environmental factors to be addressed:	MIL-STD-1472B MSFC-STD-512		
	- temperature minimum/maximum	Malone Bioastronautics	Yes	
	- temperature uniformity		Yes	
	- ventilation, placement of ducts		Yes	
	- humidity		Yes	
	- illuminance		Yes	
	- emergency illumination		Yes	
	- noise		Yes	
	. levels		Yes	
	. frequencies		Yes	
	. vibration		No	
	. noise attenuation		Yes	
	. communications		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Procedural Documentation

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Documentation Fidelity	o Corresp. of Doc. with opns. to be performed		Yes	
	o Corresp. of Doc. nomencl. with nomencl. on panel		Yes	
	o Corresp. of system response to Doc. (feedback of operator action completion)		Yes	
	o Task sequence based on task analysis		Yes	
Information Accessibility	o Physical location of Doc.		Yes	
	o Volume Organization		Yes	
	o Volume Labeling		Yes	
	o Tables of Contents Organization		Yes	
	o Contents Organization		Yes	
	o Sectional Identification Marking		Yes	
	o Procedural Identification Marking		Yes	
	o Step Identification Marking		Yes	
	o Binding		Yes	
Document Legibility	o Print Font	Van Cott and Kinkade	Yes	
	o Print Size	McCormick	Yes	
	o Contrast	Payne	Yes	
	o Column Separation	Kinney and Showman	Yes	
	o Strokewidth	Erdmann	Yes	
	o Width-Height Ratio	Bell	Yes	
	o Letter Spacing		Yes	
	o Word Spacing		Yes	
	o Case		Yes	
	o Lighting		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Procedural Documentation

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Documentation Readability	o Brevity	Van Cott and Kinkade Siegel Williams and Siegel Ta Lin Liau Coke Brown McCormick	Yes	
	o Memory Demand		Yes	
	o Morpheme Understanding		Yes	
	o No. of Transforms Required		Yes	
	o Position of clauses in sentence		Yes	
	o Vocabulary Diversity		Yes	
	o Word linkage		Yes	
	o Memory required for Semantic units		Yes	
	o Use of abbreviations		Yes	
	o Reasoning demands on reader		Yes	
	o Use of examples		No	
	o Use of mnemonic devices and memory aids		Yes	
	o Redundancy		Yes	
	o Level of detail in Figures and diagrams		No	
	o Word length		Yes	
	o Sentence length		Yes	
	o Density of 1-syllable words		Yes	
	o Density of Coordinate conjunctives		Yes	
	o Pictoral Instructions		Yes	
	o Task-induced processing		Yes	
	o Emphasis		Yes	
	o Leading		Yes	
	o Column size		Yes	
	o Table/Figure Design		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Procedural Documentation

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Document Usability	o Demand on short-term memory	Elliott and Joyce Folley Chenzoff	Yes	
	o Demand on long-term memory		Yes	
	o Time from reading to performing		Yes	
	o Intervening activities between reading and performance		Yes	
	o Availability of performance feedback		Yes	
	o "Reward" for implementing multiple procedures without reading (from memory)		Yes	
	o Proceduralized design of Job Aid		Yes	
	o Dual Track Presentation		Yes	
	o Tasks between performing and returning to procedure		Yes	
	o Time between performing and returning to procedure		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Anthropometry

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
General	o Equipment is designed to accommodate a specified percentage of the potential user population	MIL-STD-1472B Van Cott and Kinkade	Yes	
	o Anthropometric considerations entered to task characteristics, such as task frequency, difficulty, equipment interactions, task mobility requirements, and safety issues such as emergency egress	MIL-STD-1472B NASA MSFC-STD-512	Yes	
Anthropometric Data	o Basic body dimensions considered as part of design include, for studies body positions:	MIL-STD-1472B Van Cott and Kinkade NASA MSFC-STD-512	Yes	
	- stature		Yes	
	- weight		No	
	- eye height		Yes	
	- shoulder height		No	
	- chest height		No	
	- elbow height		No	
	- fingertip height		No	
	- waist height		No	
	- crotch height		No	
	- gluteal furrow height		No	
	- kneecap height		No	
	- calf height		No	
	- functional reach		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Anthropometry

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Anthropometric Data (continued)	o For seated body position:		Yes	
	- vertical arm reach		No	
	- sitting height		Yes	
	- eye height		No	
	- mid-shoulder height		No	
	- shoulder-elbow length		Yes	
	- elbow-grip length		Yes	
	- elbow-finger length		Yes	
	- elbow rest height		Yes	
	- thigh clearance height		Yes	
	- knee height, sitting		No	
	- popliteal height		No	
	- buttock-knee length		No	
	- buttock-popliteal length		No	
	- buttock-heel length		No	
	o Other anthropometric considerations of design impact, depending on system considerations, include:			
	- depth dimensions		No	
	- breadth dimensions		No	
	- circumferential dimensions		No	
	- surface dimensions		No	
	- hand dimensions		No	
	- foot dimensions		No	
	- head dimensions		No	
	- face dimensions		No	
	- moments of inertia		No	
	- range of joint motion		No	
	- strength		No	
	- age		No	
	- sex		No	
	- voice		No	
	- fatigue		No	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Anthropometry

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Use of Anthropometric Data	Anthropometric data are used as part of design of the following:	MIL-STD-1472B NASA Van Cott and Kinkade AFSC DH 1-3 MSFC-STD-512		
	- access dimensions, passageways, escape routes, etc.		Yes	
	- limiting dimensions, such as maximum reaching distances, control access, etc.		Yes	
	- adjustable dimensions, such as controls, seats, belts, etc.		No	
	- personnel protection equipment design/selection		Yes	
	- workspace design, such as console dimensions, reach height, and so on		Yes	

*List of References is attached.

APPENDIX P

HUMAN ENGINEERING ASPECTS/CRITERIA COMPARISON

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Procedural Documentation

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Document Usability	o Demand on short-term memory	Elliott and Joyce Folley Chenzoff	Yes	IEE603-O-1
	o Demand on long-term memory		Yes	IEE-603-O-1
	o Time from reading to performing		Yes	
	o Intervening activities between reading and performance		Yes	
	o Availability of performance feedback		Yes	
	o "Reward" for implementing multiple procedures without reading (from memory)		Yes	
	o Proceduralized design of Job Aid		Yes	
	o Dual Track Presentation		Yes	
	o Tasks between performing and returning to procedure		Yes	
	o Time between performing and returning to procedure		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: General HFE in Systems

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Fail Safe Design	Providing a fail safe design for those areas where human error and/or equipment malfunctions may have catastrophic effects on system operability	MIL-STD-1472B	Yes	IEE603-I-4**
Simplicity of Design	Providing as simple an operational design as possible, consistent with system functional requirements	MIL-STD-1472B	Yes	IEE603-O-1 IEE-603-O-6 RG1.62-OG-2 RG1.47-IG-5
Safety Design	Minimizing potential of human error during system operation and maintenance	MIL-STD-1472B Chapanis Van Cott and Kinkade	Yes	IEE603-O-1 IEE603-O-6 RG1.62-OG-2
User Acceptance	Enhancing user confidence and acceptance	AFSC DH 1-3	Yes	
Training Requirement Reduction	Training requirements reduced through simplicity of design	AFSC DH 1-3	• Yes	
Operator Performance	Minimizing human error along the dimensions of:	Malone Van Cott and Kinkade		ANS51.1-O-2 SG11-IC-1
	o Time		Yes	
	o Motor responses		Yes	
	o Decisions		Yes	

*List of References is attached.

**IEEE 603 is a trial use Standard and contains many of the items specified in IEEE 279, Criteria For Nuclear Power Plant Protective Systems (1968).

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Visual Displays

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Information Displayed	o Provides the operator with clear indications of system conditions which prompts operator actions, decisions	MIL-STD-1472B Van Cott and Kinkade AFSC DH 1-3 Chapanis McCormick	Yes	IEE603-1-6** ANS51.1-1-3
	o Specific areas to be addressed include: - content, in terms of what is to be displayed - precision required in the information displayed - information format		Yes	
	o Displayed information should not be redundantly displayed unless required at different operating stations		Yes	
	o Display failure should: - be immediately apparent to the operator - not cause a failure in the operability of the equipment associated with the display		Yes	IEE-603-1-5
	o Does not exceed operator capacity		Yes	ANS51.1-1-7

***List of References is attached.**

**IEEE 603 is a trial use Standard and contains many of the items specified in IEEE 279, Criteria For Nuclear Power Plant Protective Systems (1968).

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Visual Displays

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Location and Arrangement	Displays should be located such that:	MIL-STD-1472B Van Cott and Kinkade Malone McCormick		RG1.47-IG-1
	o They may be read in the normal operating position		Yes	
	o They require no tools to read (such as ladders, flashlights, etc.)		Yes	
	o They are oriented to the line of sight of the operator in the normal operator position		Yes	
	o Display surfaces do not reflect ambient light		Yes	
	o They are grouped according to: - usage rates - operational sequence - importance		Yes	
	o Viewing distance is accounted for in the design		Yes	
Coding	Coding should be used to facilitate:	MIL STD-1472B Van Cott and Kinkade Chapanis AFSC DH 1-3 Malone MSFC-STD-512		ANS51.1-O-1 RG1.47-IG-1
	- display discrimination		Yes	
	- identification of functionally similar displays		Yes	
	- identification of display relationships		Yes	
	- identification of critical information within a display		Yes	
	- information processing		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Visual Displays

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Display Use	o Display type selection (use) depends on the characteristics of the information to be displayed: - continues control - status monitoring - briefing/alerting - search/identification - decision making - trend analysis	MIL-STD-1472B Van Cott and Kinkade Chapanis	Yes	
General Display Characteristics to be Considered as part of CR design	o Indicator lights should not be used (in the extinguished mode) to indicate a system "go" condition	MIL-STD-1472B Van Cott and Kinkade Bioastraunautics Data Book	Yes	RG1.47-IG-1
	o These considerations include:		Yes	
	- information displayed		Yes	
	- functional grouping		Yes	
	- luminance		Yes	
	- luminance control		Yes	
	- display operability testing		Yes	
	- contrast between legends and background		Yes	
	- color coding		Yes	
	- parallax		Yes	
	- multiple legends		Yes	
	- visbility		Yes	
	- visual environment		Yes	
	- signal rate		Yes	
	- resolution		Yes	
	- discriminability		Yes	
	- legends		Yes	
	- character sizes		Yes	
	- symbology		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Visual Displays

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Display Characteristics (continued)	- mounting - test provisions - size/shape			
	o Printers - form of information presentation - take-up provisions - annotations - visibility - illumination - contrast		Yes	
Display Errors	Display design should address the following error types:	McCormick Malone Chapanis		ANS51.1-O-2
	- temporal		Yes	
	- selection (wrong display read)		Yes	
	- interpretation		Yes	
	- reading		Yes	

*List of References is attached.

INDEX OF CONTROL ROOM HFE DESIGN CONSIDERATIONS

Area: Controls

HFE Issues	Descriptions/Definitions	Reference Name*	Applicable To TMI-2 CR?	Associated Nuclear Regulations & Standards
Arrangement and Grouping	Considerations related to arrangement and grouping are as follows:	MIL-STD-1472B Van Cott and Kinkade McCormick Chapanis		RG1.62-IG-3
	o Controls grouped according to sequential relations in operation		Yes	
	o Primary controls located in most favorable position with respect to ease of reaching and operating		Yes	
	o Recurring control groups similar in layout from panel to panel		Yes	
	o Minimum/maximum control spacing addressed as part of design		Yes	
Coding	o Selection of coding methods (shape, size, color) consistent with coding requirements and other factors (ambient light, etc.)	MIL-STD-1472B Van Cott and Kinkade Bioastronautics Data Book Chapanis	Yes	
	o Coding modes (size, shape, color) consistent with system - functionally similar controls have same coding		Yes	

*List of References is attached.

REFERENCES

Reference Name	Full Reference
AFSC DH 1-3	AFSC DH 1-3, Human Factors Engineering (3rd Edition), Department of the Air Force, 1 January 1977.
Bell	Bell, G.L., Studies of Display Symbol Legibility, ESD-TR-66-316, Department of the Air Force, 1966.
Bioastronautics Data Book	Parker, J.F., and West, V.R. (eds.), Bioastronautics Data Book (2nd Ed.), NASA SP-3006, Scientific and Technical Information Office, NASA, 1973.
Brown	Brown, L.A., The Effect of Isolation, Readability, and Paragraph Organization on Learning from Written Instructional Materials. Ph.D. Dissertation, University of Nebraska, 1974.
Chapanis	Chapanis, A., Man/Machine Engineering, Brooks/Cole, 1965.
Chenzoff	Chenzoff, A.P., et al., Guidance and Specification for the Preparation of Full Proceduralized Job Aids for Organizational and Intermediate Maintenance of Electronic Subsystems. AFHRL-TR-71-23. Department of the Air Force, 1971.
Coke	Coke, E.U., Reading Rate, Readability, and Variations in Task-Induced Processing. J. of Educational Psychology, 68, (2), 1976.
Elliott and Joyce	Elliott, T.K., and Joyce, R.P., An Experimental Comparison of Proceduraland Conventional Electronic Troubleshooting, AFHRL-TR-68-1, Department of the Air Force, 1968.
Erdmann	Erdmann, R.L., and Ncal, A.S., Word Legibility as a Function of Word Size, Word Familiarity, and Resolution of Parameters, J. of Applied Psychology, 52, (5), 1968.
Folley	Folley, J.D., et al., Full Proceduralized Job Performance Aids, AFHRL-TR-71-53, Department of the Air Force, 1971.
Kinney and Showman	Kinney, G.C., and Showman, D.J., Studies in Display Symbol Legibility, ESD-TR-67-106, Department of the Air Force, 1967.

Malone	Malone, T.B. and Shenk, S., Human Engineering Data Guide for Evaluation, Essex Corp., 1977.
McCormick	McCormick, E.J., Human Factors Engineering, McGraw-Hill, 1970.
MIL-STD-1472B	Military Standard, Human Engineering Design Criteria for Military Systems, Equipment and Facilities. Department of Defense. 31 December 1978.
MSFC-STD-512	MSFC-STD-512 Man-System Design Criteria for Manned Orbiting Payloads, Marshall Space Flight Center, NASA, 1974.
NASA	NASA Reference Publication 1024, Anthropometric Source Book, Vol. I, Anthropometry for Designers, NASA, 1978.
Payne	Payne, D.E., Readability of Typewritten Material: Proportional vs. Standard Spacing.
Siegal	Siegal, A.I., et al., Techniques for Making Written Material More Readable/Comprehensible, AFHRL-TR-74-47, Department of the Air Force, 1974.
Ta Liu Liau	Ta Liu Liau, et al., Modification of the Coleman Readability Formulas. J. of Reading Behavior, VIII, (4), 1976.
Van Cott and Kinkade	Van Cott, H.P. and Kinkade, R.G., Human Engineering Guide to Equipment Design (Revised Edition), Joint Army-Navy-Air Force Steering Committee, 1972.
Williams and Siegel	Williams, A.R., Siegal, A.I. and Burkett, J.R., Readability of Textual Materials — A Survey of the Literature, AFHRL-TR-74-29, Department of the Air Force, 1974.

APPENDIX Q

DESIGN BASES

SURVEY OF DESIGN BASES

Categories	Calvert Cliffs-1	Three Mile Island-2	Oconee-3
Anthropometry	o U.S. Military Standards (reported 4)	o 5 ft. 6 in. to 5 ft. 9 in. or 6 ft. 4 in.(?) C/D 30 in. to 7 ft. from floor (reported 1 and 2)	o 5 ft. 2 in. to 6 ft. 2 in./male and female walk-through tested for reach envelope, visibility and traffic patterns (reported 3)
Procedures	o Operator input (reported 4) o Engineering consultation/advice (reported 4) o Test result (reported 4) o ANSI N18.7	o Operator input (TMI-1) (reported 7) o Engineering consultation/advice (reported 7) o Test Results (reported 7) o ANSI N18.7 o Two operators perform togehter on each procedure; or procedure is performed by one operator using one hand	o Operator input (reported 3) o ANSI N18.7 o Design Engineering tests and simulations (reported 3) o Review by Technical Specialists (Systems) (reported 3) o Independent review by staff (HQ) specialists (reported 3)
Data Entry Devices	o Some BOP measurements will not be displayed directly on the panel boards o Alarm history must be available throughout control room o Operator must be able to alter alarm display function/format quickly with minimum chance of error o Trend data must be kept automatically on a wide variety of plant systems o Provide logging service to operator (reported 10)	o If space to display elsewhere will <u>not</u> use computer (reported 7) - to allow continuous surveillance - to make plant safely operable without the computer o Alarm history K/B call up o Trend data K/B call up	o K/B call up of CRT Displays — parameters and "canned" display formats o K/B call up of EP's on a slide projection screen o "Item Entry" K/B call up available for predefined data (single key call up)

SURVEY OF DESIGN BASES (CONT'D.)

Categories	Calvert Cliffs-1	Three Mile Island-2	Oconee-3
Labeling	o Utility's Standard Abbreviations List (reported 6) o Red indicates warning (for as-built labels) o Operator training/experiece will enable the selection of the correct label nearby a panel component o One standard size and font (not including component engravings) o Operators will not need to read labels at a distance of greater than 9 or 10 ft. o Large percentage of labels would be operator backfits	o A-E Standard Abbreviation List (reported 2) o White on black contrast (reported 2) o Letter size specified (reported 1) o Unambiguous and not obscured by operator actions (reported 2) o Positioned over control (reported 2) o Did not duplicate reading conditions in Control Room (reported 1 and 2)	o Utility's experience and "standards" from design engineering o Some vendor supplied standard labels o Operator backfits with engineering approval — these are logged for future facility designs o By and large done through an iterative process between design personnel and plant personnel
Display Selection	o Readability at required distances (reported 4 and 5) o Size (reported 5) o Qualification (reported 5) o Integrated alarms (reported 6) o Past experience (Annunciators) (reported 6) o Redundant vrs/audio for alarm displays	o Fossil experience (reported 2) o Nuclear vendor's recommendations (reported 2) o Save space (small) (reported 2) o Ruggedness and maintainability (reported 1 and 2) o Readability — "live zero" meters (reported 2)	o Based on T&E in the utility's instrumentation section. Tests on: - size - quality - reliability - historical performance - data availability - readability

SURVEY OF DESIGN BASES (CONT'D.)

Categories	Calvert Cliffs-1	Three Mile Island-2	Oconee-3
Control/Display Grouping	o System (reported 4) o Importance (most important C/D in middle of panel section) (reported 4 and 6) o Minimize wiring (reported 5) o Mockup evaluation (reported 4, 5, and 6) o Frequency of use (reported 6) o Controls within easy reach of operator (requirement for redundancy and separation) (reported 10)	o A-E judgment of whether system needed graphics (reported 1) - graphic or mimic - semigraphic - nongraphic (most panel representation) o Criticality (reported 2) o Logical flow by system (reported 2) o Avoid mirror imaging (reported 2) o Save space but preserve operability (reported 2) o Conventions or rules for grouping - mimicking (reported 2) - functional (reported 2 and 8) - group laterally by type of control or display (all 2 position discrete rotary switches at same level on a panel) (reported 2) - frequency of use (reported 1) - tried to physically locate display near control (reported 1)	o By system/subsystem (reported 3) o By function/operations (reported 3) o Based on simulations and walk-throughs
Switch Orientation	o SBM design (reported 4) o CMC design o Mimic conventions	o Industry stds. (reported 2) o Mimic conventions (reported 2) o Toggle switches (reported 2) - up = on - down = off o Clockwise type controls (reported 2) - right = open/on - left = trip - up = off	o Generally in columns with guage readouts above switches in control room o N/S orientation within column o On/off simple switches (reported 3) - right = on - left = off

SURVEY OF DESIGN BASES (CONT'D.)

Categories	Calvert Cliffs-1	Three Mile Island-2	Oconee-3
Use of Mimicking	o Straight forward (clarity) (reported 5) o System used infrequently (reported 5) o Where physically possible used (reported 5)	o Give operator good grasp of his power flow configurations (reported 2) o Only with electrical power flow as take up too much space to mimic (reported 2)	o Only used twice: 1. original design for turbine 2. backfit feedwater by operators
Control Room Layout	o Previous Nuclear Design experience (reported 6) o Mockup evaluation (reported 6) o Operator preference (reported 6) o Size of mimic panels (reported 6) o Possibility of inadvertent actuation (reported 6) o Preliminary operator procedures (reported 6) o Detroit Edison Nuclear experience (reported 6) o Two units controlled from one room (reported 6)	o Panels arranged to allow ready accessibility to most frequently used controls (reported 8) o Controls and displays grouped on panels according to function (reported 8)	o Previous Fossil experience o Previous Nuclear experience o Operator inputs o Mockup evaluation o Simulation test results o Design Engineering inputs
Basis for Automating Actions	o Frequency of action (reported 6) o Required immediate response (reported 6) o On-line continuously	o SFAS for safety (immediacy of response) (reported 8) o To cut down on operator's operations (frequency of action) (reported 8)	o Frequency of action o Immediacy of response

SURVEY OF DESIGN BASES (CONT'D.)

Categories	Calvert Cliffs-1	Three Mile Island-2	Oconee-3
Basis for Distributing Systems Between Primary and Satellite Panels	o Less importance to plant operation (reported 6) o Frequency of use o Time available to respond to failures	o Separate protection from control instrumentation (FSAR) o Frequency of use	o No control readouts on satellites o Not primarily used o Distribution Systems — busses o Redundant features
Panel Color	o Contrast with displays (reported 6) o Lighting study (reported 6)	o Contrast with TMI-1 (reported 1) o Looked at swatches and chose color that would contrast well with normal black switches (reported 2)	o Lighting study (panel is sand blasted STN/STL) (reported 3) o VB are dark brown — contrast lights o Mockup evaluation (reported 3)
Lighting	o Recommendations of Utility lighting consultant (reported 6) o Detroit Edison experience (reported 6)	o A-E criteria (reported 7) - 160 ft. candles controllable by switches (operator controlled banks of lights) (reported 2) - level set by electrical engineers (reported 1) o Lighting intensity levels are as recommended in the Illumination Engineering Society Handbook (reported 8) o Circuiting is in accordance with the National Electrical Code (reported 8) o Normal lighting system luminaries are on alternate circuits in an area so that loss of one circuit in an area does not result in loss of more than 50% of the area's illumination (reported 8)	o Simulations o Design engineering experience o Operator inputs o Illumination engineering stds.

SURVEY OF DESIGN BASES (CONT'D.)

Categories	Calvert Cliffs-1	Three Mile Island-2	Oconee-3
Lighting (cont'd.)		o Control Room and Diesel Generator Building lighting are powered from the ESF buses for reliability under normal and emergency conditions (reported 8) o Self-contained battery-operated emergency lighting units are powered from self-contained or locally mounted batteries for emergency lighting (reported 8) o Exit signs are powered from normal lighting system and from a locally mounted battery during emergency conditions (reported 8)	
Annunciator Grouping	o Over panel serviced (reported 5) o Grouped by system, subsystem, component (reported 5)	o Alignment with controls (on same panel or in direct line) (reported 2) o Most important on top level or row within a block of annunciators, no left to right grouping (reported 2)	o Grouped by system
Auditory Alarms	o Manufacturer's standards	o Usually bought with annunciators, no evaluation done (reported 1)	o Standard from vendor o Selected for discriminability

SURVEY OF DESIGN BASES (CONT'D.)

Categories	Calvert Cliffs-1	Three Mile Island-2	Oconee-3
CR Noise Level	o Architects (reported 5) o Alarms were off-the-shelf (reported 5)	o Alarms were off-the-shelf (reported 7) o Not considered in design (reported 2)	o Alarms are standard from vendor o Other bells (alarm computer) selected for discriminability o Carpets installed as absorbers
Communications System	o Precedents (reported 5) o Experience with Fossil plants o Multiple redundancy	o The normal page — party line system shall (reported 8): 1. provide communications throughout the unit 2. be compatible with the equipment of TMI Unit 1 3. provide a communications link between TMI Unit 1 and TMI Unit 2 4. provide a redundant communications arrangement with the Emergency Page — Party Line System 5. insure reliability by being powered from the vital power buses and arranging the power and sound circuiting so that any disruption of the system in the seismic Class II areas does not affect the operation of the system in the seismic Class I areas. Also, the system circuiting shall be arranged so that failure of a circuit in an area still allows partial communications in that area.	o Redundancy — and then some - phones - sound PWR - radio - P.A. o Emergency power — voice operation o Prior Fossil experience

Categories	Calvert Cliffs-1	Three Mile Island-2	Oconee-3
Communications System (cont'd.)		o The Emergency Page — Party Line System shall provide a redundant communications system for the orderly emergency shutdown of the unit in the event that the Normal Page — Party Line System is inoperative (reported 8) o The Maintenance Telephone System shall provide communication for the testing and maintenance of the instrumentation systems (reported 8) o The Commercial Telephone System shall provide a communication link between the control rooms and service buildings of TMI Unit 2 and TMI Unit 1 and with offsite areas and the outside (reported 8) o The Microwave Communication System shall provide a communications link between Three Mile Island and Metropolitan Edison's main office (reported 8) o The Evacuation Alarm System shall alert personnel to radiation and fire hazards (reported 8) o The two-way radio communication system shall provide a direct communication link between TMI Unit 2 and Dauphin County Civil Defense and Commonwealth Defense, and provide a tie between TMI Unit 1 and TMI Unit 2 communications desks (reported 8)	

SURVEY OF DESIGN BASES (CONT'D.)

Categories	Calvert Cliffs-1	Three Mile Island-2	Oconee-3
Control Selection	o Fossile experience (SBM) o Info. Display (CMC) o Required for guarding o Size, ease of modifying and removal (reported 6)	o Pistol grip handle for positive actuation (reported 2) o SBM for compactness and adequacy (reported 2) o Ruggedness, ease of actuation and ease of access (reported 2) o Client preference (reported 1) o Operator preference (reported 1)	o Simple as possible on/off where ever possible (reported 3)
Maintainability	o Standardization (reported 6) o Minimization of interconnections and interwiring o Interchangeability of subunits	o RTMs, ISA stds., IEEE stds., were followed (reported 2) o Purchased rugged materials (reported 1)	o Duke (utility) investigated
Annunciator Activation	o Pre-trip conditions (reported 10)	o Pre-trip conditions (reported 8)	o Pre-trip conditions (reported 9)
No. of Operators/Shift and Role	o One operator — BG&E decision (reported 6)	o One operator (reported 8) o No formal requirements (reported 1) o Assumed 2 or 3, one with hands on controls (reported 2) o NRC Tech. specs. (reported 2)	o One operator is the basis for design (reported 3)
Color Coding Conventions	o Color of lights required by utility BG&E selected colors	o Standard power industry codes (reported 2) o Instrument Society of American color coding (reported 2) o ISA5.2 (reported 1)	o Red-open/energized (reported 3) o Green-closed/deenergized (reported 3) o Carried over from plants dating back to 1950s (reported 3)

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
IEE497-DB-1	A specific design basis for the post accident monitoring instrumentation shall be established for each nuclear power generation station. The design basis information thus provided shall be available, as needed, for making judgments on the adequacy of design of the post accident monitoring instrumentation. The methods for development of the specific design basis information are not within the scope of this document. The design basis shall document, as a minimum:	IEEE STD 497-1977		
	o The generating station postulated accidents for which post accident monitoring instrumentation is required.	IEEE STD 497-1977		
IEE497-DB-2	o The safety systems that are required to mitigate the consequences of the postulated accidents referred to in 4.1.	IEEE STD 497-1977		
IEE497-DB-3	o The required operator actions and the conditions under which these actions are required during the post accident period.	IEEE STD 497-1977		
IEE497-DB-4	o The generating station variables to be used by the operator to: (a) identify the accidents mentioned in Section 4.1 above to the degree necessary for the operator to perform his role; (b) assess the accomplishment of the safety functions performed by the systems mentioned in Section 4.2 above; (c) guide the operator in accomplishing the required actions referred to in Section 4.3 above; and (d) follow the course of the accident to determine whether or not conditions are evolving within safe limits.	IEEE STD 497-1977		
IEE497-DB-5	o The portion of the post accident monitoring instrumentation that is Class IE.	IEEE STD 497-1977		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
IEE497-DB-6	o The events or conditions or both which determine the time period during which the monitoring of each variable.	IEEE STD 497-1977		
IEE497-DB-7	o The time after the postulated accidents when each variable referred to in Section 4.4 is first required to be monitored and the time interval during which it is required to be monitored.	IEEE STD 497-1977		
IEE497-DB-8	o The minimum number and location of the sensor(s) required for any variable referred to in Section 4.4 that have a spatial dependence.	IEEE STD 497-1977		
IEE497-DB-9	o The locations at which the information must be available to the operator and the types of information (for example: discrete state, current value of a continuous variable, long term trend) which must be presented.	IEEE STD 497-1977		
IEE497-DB-10	o The range of transient and steady-state conditions of both the energy supply and the environment (for example: voltage, frequency, electromagnetic interference, temperature, humidity, pressure, vibration, and radiation) for which provisions must be incorporated to ensure adequate performance when required.	IEEE STD 497-1977		
IEE497-DB-11	o The malfunctions, accidents, or other unusual events (for example: fire, explosion, missiles, lightning, flood, earthquake, wind) which could physically damage components or could cause environmental changes leading to degradation of the performance of this instrumentation and which the design must withstand.	IEEE STD 497-1977		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
IEE497-DB-12	o The maximum and minimum values and the maximum rate of change of each variable which must be accommodated by the post accident monitoring instrumentation and the maximum error within which the information must be conveyed to the operator for all of the applicable conditions listed in 4.10 and 4.11 above.	IEEE STD 497-1977		
IEE388-DB-1	Interrelationship among the systems, components, and human factors in each phase of the test activity shall be considered and reflected in the system design and layout.	ANSI/IEEE STD 388-1977		
IEE388-DB-2	Provision shall be made for locating test equipment and access to test points to minimize the effort and time required to perform checks, inspections, functional tests, and calibration verification tests.	ANSI/IEEE STD 388-1977		
IEE388-DB-4	Testing programs shall be conducted in a logical sequence such that the overall condition of the systems under test can immediately be assessed and the need for progressing further into the testing of individual components be determined.	ANSI/IEEE STD 388-1977		
IEE388-DB-5	The test program of each system shall be designed to provide for minimum interference with related operational channels, systems, or equipment.	ANSI/IEEE STD 388-1977		
	o General. The design bases for the control and display facilities in the control room should be established and documented, before beginning the detailed control room design, and updated as needed.	IEEE STD 566-1977		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
	o Contents. The design bases should include but not be limited to the following items:			
IEE566-DB-1	- The operating modes for which the central control room display and control facilities should be designed.	IEEE STD 566-1977		
IEE566-DB-2	- The number of operators and the responsibilities assigned to them under each operating mode.	IEEE STD 566-1977		
IEE566-DB-3	- The functional areas into which the control room is to be organized. These may include the normal, emergency, and supporting operations areas.	IEEE STD 566-1977		
IEE566-DB-4	- The basis for grouping of display and control devices within any functional area (See Section 6.)	IEEE STD 566-1977		
IEE566-DB-5	- The limiting number of display devices which can be active at the same time, by type, established as a design goal for each functional area of the control room to avoid operator sensory saturation. (See Appendix B.)	IEEE STD 566-1977		
IEE566-DB-6	- A listing and classification of the safety related display and control instrumentation and any post accident monitoring instrument for which specific requirements are already established by regulatory requirements, industry standards, or safety analysis reports. (See Ref [1], [2].)	IEEE STD 566-1977		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
IEE566-DB-7	- The requirements which are mandated by, or directed by, user company policies or contracts or both.	IEEE STD 566-1977		
IEE566-DB-8	- The anthropometric relationship to be used for design of the control boards.	IEEE STD 566-1977		
IEE566-DB-9	- The list of functions, the controls for which may be transferred from the central control room facilities to remote facilities.	IEEE STD 566-1977		
IEE566-DB-10	- The sequence of events for the postulated design basis events.	IEEE STD 566-1977		
IEE566-DB-11	- Data to be used for trend and historical record purposes.	IEEE STD 566-1977		
IEE279-DB-1	A specific protection system design basis shall be provided for each nuclear power plant. The information thus provided shall be available, as needed, for making judgments on system functional adequacy. The design basis shall document as a minimum, the following: (a) the plant conditions which require protective action; (b) the plant variables (e.g., neutron flux, coolant flow, pressure, etc.) that are required to be monitored in order to provide protective actions; (c) the minimum number and location of the sensors required to monitor adequately, for protective function purposes, those plant variables listed in 3(b) that have a spatial dependence; (d) prudent operational limits for each variable listed in 3(b) in each applicable reactor operation mode;	IEEE 279-1968		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
	(e) the margin, with appropriate interpretive information, between each operational limit and the level considered to mark the onset of unsafe conditions; (f) the levels that, when reached, will require protective system action; (g) the range of transient and steady-state conditions of both the energy supply and the environment (e.g., voltage, frequency, temperature, humidity, pressure, vibration, etc.) during normal, abnormal, and accident circumstances throughout which the system must perform; (h) the malfunctions, accidents, or other unusual events (e.g., fire, explosion, missiles, lightning, flood, earthquake, wind, etc.) which could physically damage protection system components or could cause environmental changes leading to functional degradation of system performance, and for which provisions must be incorporated to retain necessary protection system action; (i) minimum performance requirements including the following: 1) system response time; 2) system accuracies; 3) ranges (normal, abnormal and accident conditions) of the magnitudes and rates of change of sensed variables to be accommodated until proper conclusion of the protection system action is assured.			

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
IEE279-DB-2	System Repair. The system shall be designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.	IEEE279-1968		
IEE308-DB-1	Controls. Automatic and manual controls shall be provided to: (1) Select the most suitable power supply to the distribution system. (2) Disconnect appropriate loads when the preferred power supply is not available. (3) Start and load the standby power supply. Manual controls shall be provided to permit the operator to select the most suitable distribution path from the power supply to the load.	IEEE Std 308-1971		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
IEE603-DB-1	A specific basis[3] shall be established for the design of the safety system of each nuclear power generating station. The design basis shall also be available as needed to facilitate the determination of the adequacy of the safety system, including design changes. The design basis shall document, as a minimum: o o o 3.5 Those protective actions, identified in 3.2, that may be initiated solely by manual means, and shall document for each: 3.5.1 The justification for permitting manual initiation 3.5.2 The variables to be monitored to facilitate the manual initiation of protection action 3.5.3 The minimum performance requirements including the following for the appropriate combinations of those conditions of 3.7 and 3.8: 3.5.3.1 System response times with appropriate interpretive information 3.5.3.2 System accuracies 3.5.4 The range of environmental conditions imposed upon the operator during normal, abnormal, and accident circumstances throughout which the manual operations must be performed	IEEE STD 603-1977		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
	Specific control room design bases to be established include:			
IEEP567-DB-1	o Seismic considerations	Draft IEEE STD P567/4H		
IEEP567-DB-2	o Radiation shielding	Draft IEEE STD P567/4H		
IEEP567-DB-3	o Natural and other phenomena	Draft IEEE STD P567/4H		
IEEP567-DB-4	o Missiles	Draft IEEE STD P567/4H		
IEEP567-DB-5	o Noise Sources	Draft IEEE STD P567/4H		
IEEP567-DB-6	o Piping	Draft IEEE STD P567/4H		
ANS56.3-DB-1	The testing requirements are intended to accomplish a combination of the objectives listed below: (1) Capability to reliably perform its intended safety function (2) Operability over the design service life (3) Detection of degrading conditions 5.1.2. The testing requirements are limited to those associated with pre-operational, start-up and operational testing to periodically assess and verify the overpressure protection capability.	ANSI/ANS-56.3-1977 (N193)		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
CFR-DB-1	Criterion 1 -- Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.	10 CFR Part 50		
CFR-DB-2	Criterion 2 -- Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.	10 CFR Part 50		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
CFR-DB-3	Criterion 3 -- Fire protection. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.	10 CFR Part 50		
CFR-DB-4	Criterion 4 -- Environmental and missile design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.	10 CFR Part 50		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
CFR-DB-5	Criterion 5 -- Sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.	10 CFR Part 50		
CFR-DB-6	Criterion 10 -- Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	10 CFR Part 50		
CFR-DB-7	Criterion 13 -- Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	10 CFR Part 50		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
CFR-DB-8	Criterion 15 -- Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	10 CFR Part 50		
CFR-DB-9	Criterion 19 -- Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.	10 CFR Part 50		
CFR-DB-10	Criterion 30 -- Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.	10 CFR Part 50		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
CFR-DB-11	Criterion 32 -- Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.	10 CFR Part 50		
CFR-DB-12	Criterion 24 -- Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.	10 CFR Part 50		
CFR-DB-13	Criterion 26 -- Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel	10 CFR Part 50		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
	design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.			
CFR-DB-14	Criterion 63 -- Monitoring fuel and waste storage. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.	10 CFR Part 50		
CFR-DB-15	Criterion 64 -- Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	10 CFR Part 50		
CFR-DB-16	Emergency plans shall contain, but not necessarily be limited to, the following elements: o o o C. Means for determining the magnitude of the release of radioactive materials, including criteria for determining the need for notification and participation of local and State agencies and the Atomic Energy Commission and other Federal agencies, and criteria for determining when protective measures should be considered within and outside the site boundary to protect health and safety and prevent damage to property;	10 CFR Part 50		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
CFR-DB-17	Measures shall be established to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the nuclear power plant or fuel reprocessing plant. These measures shall provide for the identification of items which have satisfactorily passed required inspections and tests, where necessary to preclude inadvertent bypassing of such inspections and tests. Measures shall also be established for indicating the operating status of structures, systems, and components of the nuclear power plant or fuel reprocessing plant, such as by tagging valves and switches, to prevent inadvertent operation.	10 CFR Part 50		
CFR-DB-18	B. The licensee shall establish an appropriate surveillance and monitoring program to: 1. Provide data on quantities of radioactive material released in liquid and gaseous effluents to assure that the provisions of paragraph A of this section are met; 2. Provide data on measurable levels of radiation and radioactive materials in the environment to evaluate the relationship between quantities of radioactive material released in effluents and resultant radiation doses to individuals from principal pathways of exposure; and 3. Identify changes in the use of unrestricted areas (e.g., for agricultural purposes) to permit modifications in monitoring programs for evaluating doses to individuals from principal pathways of exposure.	10 CFR Part 50		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
ANS2.2-DB-1	Instrumentation shall be provided depending on the plant's Safe Shutdown Earthquake maximum ground acceleration as specified below. Instruments shall be provided at the representative locations to achieve the stated purpose of this standard. Instruments shall be located where comparison can be made after an earthquake with the calculated vibratory responses used in the seismic design.	ANSI/ANS-2.2-1978		
ANS4.1-DB-1	The designers shall determine, by means of a systematic analysis, that (a) the monitored process variable can provide the required information during the Design Basis Events. (b) the equipment can perform in the configuration specified for its installation. (c) the interactions of protective actions, control actions, and the environmental changes that caused, or are caused by, the Design Basis events do not prevent the mitigation of the consequences of the event; and (d) the equipment in the configuration specified for its installation cannot easily be made inoperational by the inadvertent actions of operating or maintenance personnel.	ANSI/ANS-4.1-1978		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
ANS4.1-DB-2	The design of the safety systems and the safety supporting systems shall permit implementation of operating and maintenance procedures for the surveillance, calibration, adjustment, and repair of the protection and actuator systems without inducing a Design Basis Event or an unprotected condition. The designer shall give special consideration to preventing inadvertent modification of the systems that may negate the intent of the system design.	ANSI/ANS-4.1-1978		
ANS4.1-DB-3	3.9 Surveillance. Means for surveillance of the safety systems and the safety supporting systems shall be established. They shall be adequate to: (a) determine that the performance of the safety systems and their safety supporting sytems is within prescribed limits; (b) assure that maintenance operations have been performed correctly; (c) detect trends toward unacceptable conditions; and (d) determine that the independence of redundant or diverse systems has been maintained. (e) permit the operational capability of an instrument channel, logic channel, and an actuator channel to be demonstrated.	ANSI/ANS-4.1-1978		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
SRP-DB-1	If a seismic event comparable to a safe shutdown earthquake (SSE) occurs, it is important that the operator be able to assess the condition within the containment quickly. The proper functioning of at least one leakage detection system is essential in evaluating the seriousness of the condition within the containment in the event leakage has developed in the RCPB.	SRP 5.2.5-3		
SRP-DB-2	It is important to be able to associate a signal or indication of a departure from the normal operating conditions with a quantitative leakage flow rate. Except for flow rate or level change measurements from tanks, sumps, or pumps, signals from other leakage detection systems do not provide information readily convertible to a common denominator. Approximate relationships converting these signals to units of water flow are formulated to assist the operator in interpreting signals. The instrumentation associated with the leak detection system is reviewed by EICSB in SRP 7.5 (Ref. 4). Procedures for operator evaluation of leakage conditions are reviewed by RSB.	SRP 5.2.5-3		
SRP-DB-3	The sensitivity and response time of each (Reactor Coolant Pressure Boundary) leakage detection system employed for monitoring unidentified leakage to the containment should be adequate to detect an increase in leakage rate, or its equivalent, of one gpm in less than one hour.	SRP 5.2.5-4		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
SRP-DB-4	Instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the sump water temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked throughout the course of an accident. Recording equipment capable of following the transient should be provided.	SRP 6.2.1.1.A-3		
SRP-DB-5	Instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the sump water temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked throughout the course of an accident. Recording equipment capable of following the transient should be provided.	SRP 6.2.1.1.A		
SRP-DB-6	Instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the sump water temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked throughout the course of an accident. Recording equipment capable of following the transient should be provided.	SRP 6.2.1.1.B-4		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
SRP-DB-7	Instrumentation should be provided to monitor containment heat removal system and system component performance under normal and accident conditions. The instrumentation should be capable of determining whether a system is performing its intended function, or a system train or component is malfunctioning and should be isolated. The instrumentation should be redundant and where practical, diverse, and should have readout and alarm capability in the control room.	SRP 6.2.2-4		
SRP-DB-8	The design of the containment isolation system is acceptable if provisions are made to allow the operator in the main control room to know when to isolate by remote-manual means fluid systems that have a post-accident safety function. Such provisions may include instruments to measure flow rate, sump water level, temperature, pressure, and radiation level.	SRP 6.2.4-6		
	In general, the control room inlets must be so placed in relation to the location of potential release points as to minimize control room contamination in the event of a release. Specific criteria as to radiation and toxic gas sources are as follows:			
SRP-DB-9	<u>Radiation Sources</u> As a general rule the control room ventilation inlet should be separated from the major potential release points by at least 100 ft. laterally and by 50 ft. vertically. However, the actual minimum distances must be based on the dose analyses. Refer to Section III of this plan and Reference 7 for further information.	SRP 6.4-4		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
SRP-DB-10	Toxic Gases The minimum separation distance is dependent upon the gas in question, the container size, and the available control room protection provisions. Refer to Regulatory Guide 1.78 (Ref. 3) for general guidance and to Regulatory Guide 1.95 (Ref. 4) for specific acceptable design provisions related to chlorine.	SRP 6.4-4		
SRP-DB-11	Toxic Gases For acceptance purposes, three exposure categories are defined: protective action exposure (2 minutes or less), short-term exposure (between 2 minutes and 1 hour), and long-term exposure (1 hour or greater). Because the physiological effects can vary widely from one toxic gas to another, the following general restrictions should be used as guidance: there should be no chronic effects from exposure, and acute effects, if any, should be reversible within a short period of time (several minutes) without benefit of medication other than the use of self-contained breathing apparatus. The allowable limits should be established on the basis that the operators should be capable of carrying out their duties with a minimum of interference caused by the gas and subsequent protective measures. The limits for the three categories normally are set as follows: (1) Long-term limit (1 hour or greater): use a limit assigned for occupational exposure (40-hour week). (2) Short-term limit (2 min. to 1 hour): use a limit that will assure that the operator will not suffer incapacitating effects after a one-hour exposure.	SRP 6.4-4		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
	(3) Protective action limit (2 min. or less): use a limit that will assure that the operator will quickly recover after breathing apparatus is in place. In determining this limit, it should be assumed that the concentration increases linearly with time from zero to two minutes and that the limit is attained at two minutes. The protective action limit is used to determine the acceptability of emergency zone protection provisions during the time personnel are in the process of fitting themselves with self-contained breathing apparatus. The other limits are used to determine whether the concentrations with breathing apparatus in place are applicable. (They are also used in those cases where the toxic levels are such that emergency zone isolation without use of protective gear is sufficient.) As an example of appropriate limits, the following are the three levels for chlorine gas: Long-term: 1 ppm by volume Short-term: 4 Protective action: 15			
SRP-DB-12	The identification of safety-related systems is acceptable when it can be concluded that the integrated response of these systems assures the safety of the plant in normal operation, anticipated operational transients, and postulated accidents. o o o	SRP 7.1 (Introduction)		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
SRP-DB-13	The fundamental bases for acceptance of the proposed technical specifications are that the limiting conditions for operation are such that sufficient equipment is required to be available for operation to meet the single failure criterion; that equipment outages that are permissible for a short period of time still leave available sufficient equipment to provide the protective function assuming no failures; and that the provisions of the technical specifications are compatible with the safety analyses.	SRP 7.1 (Introduction)		
SRP-DB-14	Design Criterion 1, "Quality Standards and Records," of Appendix A of 10 CFR Part 50. General Design Criterion 1 also requires that, "Structures, systems and components important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed." Therefore, the SAR should include (1) a discussion regarding the applicability of each criterion listed, and (2) a statement to the effect that the criteria are implemented (OL) or will be implemented (CP) in the design of safety-related instrumentation and control systems.	SRP 7.1-4		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
SRP-DB-15	Automatic initiation is required for all protective functions that must be started within a short time of the indicated need for the function. Although GDC 20 appears to require automatic initiation of all protective functions, initiation solely by manual means has been acceptable. However, automatic initiation is preferable for all protective functions, even though they are not needed (according to the accident analyses) for a relatively long time. Where the protective action is initiated solely by manual means, all the actions that need or may need to be performed by the operator during the time interval are reviewed, as are the applicant's basis for not providing automatic initiation. In this latter regard, the cost of automatic initiation is not, of itself sufficient justification for using manual initiation. If the reviewer's judgment is that manual initiation is sufficiently reliable, then the equipment used by the operator to detect the need for the protection function, and to verify that the protective function has been completed, it must also meet all the requirements applicable to automatically initiated protective functions.	SRP 7.3-7		
SRP-DB-16	Test frequencies are acceptable if identical to frequencies recently approved on other identical plants. Any changes made in design or test procedure are not an adequate basis for reducing test frequencies until after experience is gained and the results submitted for review.	SRP 7.3-10		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
SRP-DB-17	The safety-related display instrumentation design is acceptable when it can be concluded that it conforms to the criteria listed in Table 7-1 and that the operator will be provided with sufficient information to perform required manual safety functions should such action be necessary. Specific points with regard to these criteria are detailed below.	SRP 7.5-2		
SRP-DB-18	The SRDI should cover appropriate variables, consistent with the assumptions for accident analyses and with the information needs of the operators in normal, transient, and accident conditions. The design of the SRDI should conform to the recommendations of Branch Technical Position EICSB 23. The accuracy and range of indicating instrumentation should be consistent with the assumptions of the accident analyses. Any exceptions to these requirements will be referred to the appropriate branch for resolution on an individual case basis.	SRP 7.5-2		
SRP-DB-19	Redundant channels of indicating instrumentation should be isolated physically and electrically to assure that a single failure will not result in complete loss of information about a monitored variable. Single failures might include such possible faults as shorting or opening circuits or interconnecting signal or power cables. It also includes single credible malfunctions or events that might cause a number of subsequent component, module, or channel failures. The post-accident SRDI should be capable of operating from onsite power. If signals from the post-accident monitoring equipment are used for control, the required isolation devices will be classified as part of the post-accident monitoring instrumentation. No credible failure at the output of an isolation device should prevent the associated monitoring channel from meeting minimum performance requirements considered in the design bases.	SRP 7.5-2		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
SRP-DB-20	Components and modules should be of a quality consistent with the reliability requirements for safety-related systems. An acceptable quality would be that of components and modules that have been previously used in similar service conditions and have demonstrated low maintenance requirements and failure rates. Other means to demonstrate acceptable quality would be through analysis and testing of components and modules, in accordance with criteria cited in Table 7-1.	SRP 7.5-2		
SRP-DB-21	The "other instrumentation systems required for safety" are acceptable when it is determined that these systems satisfy the following requirements: 1. They have the required redundancy. 2. They meet the single failure criterion. 3. They have the required capacity and reliability to perform intended safety functions on demand. 4. They are capable of functioning during and after certain design basis events such as earthquakes, accidents, and anticipated operational occurrences.	SRP 7.6-3		
SRP-DB-22	The control systems not required for safety are acceptable if failures of control system components or total systems would not significantly affect the ability of plant safety systems to function as required, or cause plant conditions more severe than those for which the plant safety systems are designed.	SRP 7.7		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
SRP-DB-23	There are no general design criteria or regulatory guides that directly apply to the safety-related performance requirements for the communication system. The APCSB will use the following criterion to assess the system design capability: the communication system is acceptable if the integrated design of the system will provide effective communication between plant personnel in all vital areas during the full spectrum of accident or incident conditions under maximum potential noise levels.	SRP 9.5.2		
SRP-DB-24	Lighting and two way voice communication are vital to safe shutdown and emergency response in the event of fire. Suitable fixed and portable emergency lighting and communication devices should be provided to satisfy these requirements.	SRP 9.5.1-32		
SRP-DB-25	An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.	SRP 15.1.1-3		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
RG145-DB-1	The safety significance of leaks from the reactor coolant pressure boundary (RCPB) can vary widely depending on the source of the leak as well as the leakage rate and duration. Therefore, the detection and monitoring of leakage of reactor coolant into the containment area is necessary. In most cases, methods for separating the leakage from an identified source from the leakage from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action should a leak be detrimental to the safety of the facility. Identified leakage is: (1) leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered, and conducted to a sump or collecting tank, or (2) leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of unidentified leakage monitoring systems or not to be from a flaw in the RCPB. Unidentified leakage is all other leakage.	Regulatory Guide 1.45		
RG168-DB-1	In the design of nuclear power plants, postulated accident assumptions are often explicitly or implicitly bounded by the design of control and instrumentation systems (e.g., pressurizer level or feedwater flow control). In such cases, operation of the instrumentation and controls over the design operating range should be performed, and the effects of limiting malfunctions or failures should be simulated to demonstrate the adequacy of design and installation and the validity of accident analysis assumptions. Tests should be conducted, as appropriate, to verify redundancy and electrical independence.	Regulatory Guide 1.68		

SURVEY OF DESIGN BASES FOR NUCLEAR POWER PLANT CONTROL ROOMS

NUCLEAR POWER PLANT:

NUMBER	DESIGN BASIS	REFERENCE(S)	APPLICABLE TO CONTROL ROOM?	REFERENCE
RG178-DB-1	14. Detection instrumentation, isolation systems, filtration equipment, air supply equipment, and protective clothing should meet the single-failure criterion. (In the case of self-contained breathing apparatus and protective clothing, this may be accomplished by supplying one extra unit for every three units required.)	Regulatory Guide 1.78		
RG197-DB-1	It is important that accident-monitoring instrumentation components and their mounts that cannot be located in other than non-Seismic Category I buildings be conservatively designed for the intended service.	Regulatory Guide 1.97		

APPENDIX R

PHILOSOPHIES/PRINCIPLES

APPENDIX D
LIST OF HUMAN ENGINEERING PHILOSOPHIES & PRINCIPLES USED IN TMI-2 DESIGN

- PHILOSOPHY 1 — MAXIMIZE THE INFORMATION ON PLANT OPERATIONS IMMEDIATELY AVAILABLE TO THE CONTROL ROOM OPERATOR

  Principle 1a. Size the control room and control panels such that all controls and displays will be within the field of view of the operator at the Plant Control Station

  Principle 1b. Color code indicator lights

  Principle 1c. Group annuciators by systems

  Principle 1d. Display relatively slow changes in status (chart recordings)

  Principle 1e. Set absolute limits of displays to reflect the expected operational limits of the subsystem

  Principle 1f. Display on computer only if panel space is not available

- PHILOSOPHY 2 — MINIMIZE THE TIME REQUIRED TO LOCATE CONTROLS AND DISPLAYS

  Principle 2a. Place controls and displays for the same system on the same panel

  Principle 2b. Organize systems on inner benchboards by frequency of use

  Principle 2c. Organize outer vertical panels so that displays support the operator using the benchboards

  Principle 2d. Arrange controls/displays in mimic or functional groups

  Principle 2e. Locate labels in standard position with respect to subject control/display

- PHILOSOPHY 3 — MINIMIZE TIME TO RESPOND TO ALARMS

  Principle 3a. Locate critical controls/displays

  Principle 3b. Organize systems on benchboards by criticality of system

  Principle 3c. Arrange annuciators above the controls and displays for the systems they monitor

- PHILOSOPHY 4 — MAXIMIZE THE RELIABILITY OF CONTROL ROOM SYSTEMS

  Principle 4a. Use high reliability components

  Principle 4b. Minimize devices intervening between controls and devices being controlled

  Principle 4c. Minimize devices intervening between sensors and displays

APPENDIX S

INTERVIEW QUESTIONS

APPENDIX

HUMAN FACTORS QUESTIONS ON CONTROL ROOM DESIGN FOR THE UTILITY OWNING THE POWER PLANT

1. To the best of your recollection, how were the A&E and reactor manufacturer selected?
   a. Were previous control room designs reviewed during the process?
   b. Were control room operators of their panels interviewed during the process?
   c. Was the AEC contacted to determine if their panels had ever been involved in reported problems?
2. Would you please describe, to the best of your ability, the sequence of important events that led up to the installation of the control panel.
3. To the best your knowledge, did the utility place any requirements (e.g., criteria, standards) on control panel design?
   a. What were the requirements?
   b. Did the design reflect the requirements?
4. Did the utility constrain or alter in any way the design of the control panel?
   a. Panel arrangement, overall layout, organization of switches or displays?
   b. Cost, schedule?
5. Did the utility ever hold formal management reviews of the control panel prior to its being manufactured?
   a. What factors were considered important in the reviews?
   b. Did the reviews result in changes to the design? What changes?
6. Did the utility ever perform a detailed review of the panel operations?
7. To the best of your knowledge, how were the operator procedures defined and then developed into the manuals used by operators today?
8. Were the operator procedures modeled after those of another plant or plants? What were the bases for the format, organization and language of the procedures?
9. Do you know of any tests conducted to verify that the operator procedures would supply sufficient information and guidance during emergency conditions?
10. Does the utility conduct any program to identify problems in operating the control panel, or to solicit operator recommendations on potential backfits?
11. How was it procedurally determined that, for usual operating situations, one operator would be responsible for monitoring the control panels?

12. During the late 60's and early 70's, did the utility request comments from control room operators concerning the panel design?

13. What documentation did the utility require the A&E to deliver to support the control panel design: Was the A&E required to produce any documentation demonstrating the operability of the control panel? Did the utility require a specific set of design bases for control panel design?

14. Do you know of any utility personnel monitoring the control panel development that had experience in the development of other complex control rooms? Did any have human engineering training or experience?

15. Did the utility examine training problems during control room design?

16. Did the utility examine potential or real control room problems during testing or training? If yes, what data were collected; what problems uncovered; what changes made?

17. Did the utility ever conduct walk-throughs, using mockups or simulations to evaluate operator performance in using the panels. If yes, what measures were taken; what problems were uncovered; what changes were made?

17a. Bases?

18. Who manufactured the control panels?

19. Who participated in test and installation?

20. Who laid out the CR arrangement?

APPENDIX

HUMAN FACTORS QUESTIONS ON CONTROL ROOM DESIGN FOR THE REACTOR MANUFACTURER

1. To the best of your recollection, how was the reactor manufacturer selected?
   a. Were previous control room designs reviewed during the process?
   b. Were control room operators of the manufacturer's panel interviewed during the process?
   c. Was the AEC contacted to determine if the manufacturer's panels had ever been involved in reported problems?

2. Beginning with reactor manufacturer selection, would you please describe, to the best of your ability, the sequence of important events that led up to the installation of the control panel.

3. To the best of your knowledge, did the utility place any requirements (e.g., criteria, standards) on control panel design?
   a. What were the requirements?
   b. Did the design reflect the requirements?

4. Did the utility or A&E constrain or alter in any way the design of the control panel?
   a. Panel arrangement, overall layout, organization of switches or displays?
   b. Cost, schedule?

5. Were regular management reviews of the control panel concept held prior to its being manufactured?
   a. What factors were considered important in the reviews?
   b. Did the reviews result in changes to the design? What changes?

6. Was a detailed review of panel operations ever performed?

7. During the design process were alternative panel configurations taken into consideration? What were the principal factors used in selecting the final configuration?

8. Was the selected configuration similar to the one or more panels designed in the past?

9. What were the factors considered in:
   - o Control Selection
   - o Display Selection
   - o Mimicking
   - o Automatic Shutdown

10. How was it procedurally determined that, for usual operating situations, one operator would be responsible for monitoring the control panels?

11. Were comments from control room operators concerning the panel design ever requested during early design phases? Did the reactor manufacturer incorporate changes from these comments?

12. What documentation was the reactor manufacturer required to deliver to support the control panel design? Was the reactor manufacturer required to produce any documentation demonstrating the operability of the control panel? Did the utility require a specific set of design bases for control panel design?

13. Do you know of any reactor manufacturer personnel developing the control panel design that had experience in the development of other complex control rooms? Did any have human engineering training or experience?

14. Did the reactor manufacturer examine potential training problems during control room design?

15. Were features included on the control panel expressly to protect specific (expensive) equipment items from damage? If yes, what features?

16. What role did precedent play in CR Design?

    - In panel layout and arrangement?
    - In selecting manual tasks?
    - Component selection?
    - Nomenclature, marking, labeling
    - Operational strategy

17. Would you characterize the panel design approach as directed towards minimizing the likelihood of operator errors? If so, what steps were taken?

18. What acceptance tests or checks were used to assure that the as-built and delivered control room was in agreement with the reactor manufacturer's specifications?

19. What anthropometric percentile or range of percentiles were assumed for the operator?

20. What was the basis for the choice of anthropometric percentiles?

21. What conventions were used for color coding?

22. What was the basis for color coding conventions?

23. Was control panel color specified by the reactor manufacturer? If so, was contrast between displays and their background evaluated before selecting the panel color?

24. Was readability of displays at the precedurally required distances evaluated before display selection?

25. What conventions or rules were used for labeling (e.g., contents, type size, font, etc.)? Were these consistently applied?

26. What was the basis for labeling conventions or rules?

27. Was the readability of labels at procedurally required distances evaluated before final selection of label characteristics?

28. What conventions or rules were used to group controls and associated displays? Were these applied consistently?

29. What was the basis for control/display grouping?

30. What is the relationship?

31. What conventions or rules were used for the orientation of switch positions (e.g., up=on; down=off)? Were these rules on conventions followed consistently?

32. What was the basis for switch orientation conventions or rules?

33. Was design consideration given to panel operations when the operator is wearing a breathing apparatus and/or protective garments?

34. Was consideration ever given to how much information the operator must be able to correctly recall in order to operate the panel?

35. Was consideration ever given to how much information the operator must process correctly to operate the panel?

36. In selecting panel components was any consideration given to their maintainability (e.g., replacing light bulbs, changing labels, replacing switches)?

37. Was operator response time required by failures taken into account in the location of various components?

38. What AEC regulations and industry standards were used to guide the panel design?

39. Did the reactor manufacturer participate with the A&E in defining the annunciator and alarm philosophy and system? If yes, what is the philosophy and why was it chosen?

40. How was redundancy assured for the class IE displays? For the class IE controls?

41. Were walk-throughs, using mockups, or simulations ever performed to measure or observe operator performance? If so, what measures were taken; what, if any, problems were uncovered; and what changes were made?

42. Were operator performance data collected during plant and control room testing? If yes, what data; what problems were uncovered; and what changes were made?

43. Does the reactor manufacturer have a program to monitor operator performance or design comments on a continuing basis? If so, what problems have been found, and what backfits made?

44. Did the reactor manufacturer participate in developing procedures? if so, were walk-throughs/simulations used? Did the operator participate? What bases were used for the format, language and organization of the procedures?

45. Did the reactor manufacturer ever prepare detailed task analyses of operator tasks as a means to locate specific operational problems?

APPENDIX

HUMAN FACTORS QUESTIONS ON CONTROL ROOM DESIGN FOR THE ARCHITECT — ENGINEER

1. Which panels in the Control Room were not designed by the A&E?

2. Did the A&E consult, advise, assist or in other ways help with the design of the remaining panels?

3. Once the panel design was frozen, what was the policy on changes? What was the procedure on making changes? Who approved changes?

4. Who were the engineers in charge of this project from the beginning through the licensing of the plant?

5. To the best of your recollection, how was the A&E selected?
   a. Were previous control room designs reviewed during the process?
   b. Were control room operators of the A&E's panels interviewed during the process?
   c. Was the AEC contacted to determine if the A&E's panels had ever been involved in reported problems?

6. Beginning with A&E selection, would you please describe, to the best of your ability, the sequence of important events that led up to the installation of the control panel.

7. To the best of your knowledge, did the utility place any requirements (e.g., criteria, standards) on the control panel design?
   a. What were the requirements?
   b. Did the design reflect the requirements?

8. Did the utility constrain or alter in any way the design of the control panel?
   a. Panel arrangement, overall layout, organization of switches or displays?
   b. Cost, schedule?

9. Were regular management reviews of the control panel concept held prior to its being manufactured?
   a. What factors were considered important in the reviews?
   b. Did the reviews result in changes to the design? What changes?

10. Was a detailed review of panel operations ever performed?

11. During the design process were alternative panel configurations or concepts taken into consideration? What were the principle factors used in selecting the final configuration?

12. Was the selected configuration similar to one or more panels designed by Bechtel in the past?

13. What were the factors considered in:

    - Control Selection
    - Display Selection
    - Mimicking

14. How was it procedurally determined that, for usual operating situations, one operator would be responsible for monitoring the control panels?

15. Were comments from control room operators concerning the panel design ever requested during early design phases? Did the A&E incorporate changes from these comments?

16. What documentation was the A&E required to deliver to support the control panel design? Was the A&E required to produce any documentation demonstrating the operability of the control panel? Did the utility require a specific set of design bases for control panel design?

17. Do you know of any A&E personnel developing the control panel design that had experience in the development of other complex control rooms? Did any have human engineering training or experience?

18. Did the A&E examine potential personnel selection or training problems during control room design?

19. What role did precedent play in CR Design?

    - In panel layout and arrangement?
    - In selecting manual tasks?
    - Component selection?
    - Nomenclature, marking, labeling?
    - Operational strategy?
    - Automation?
    - Annunciators?

20. Would you characterize the panel design approach as directed towards minimizing the likelihood of operator errors? If so, what steps were taken?

21. What acceptance tests or checks were used to assure that the as-built and delivered control room was in agreement with the A&E specifications?

22. What is the alarm philosophy and strategy used in Calvert Cliffs? Why was it selected? Was any consideration given to prioritizing alarms? Why was it rejected?

23. What systems are automated-actions; why were these automated?

24. What use was made of video displays, and why?

25. What systems are <u>not</u> located in the primary control room? Why?

26. What anthropometric percentile or range of percentiles were assumed for the operator?

27. What was the basis for the choice of anthropometric percentiles?

28. What conventions were used for color coding?

29. What was the basis for color coding conventions?

30. Was contrast between displays and their background evaluated before selecting the panel color?

31. Was readability of displays at the procedurally required distances evaluated before display selection?

32. What bases or standards were used for control room lighting? Was lighting intended to be controlled by the operator?

33. What conventions or rules were used for labeling (e.g., contents, type size, font, etc.)? Were these consistently applied?

34. What was the basis for labeling conventions or rules?

35. Was the readability of labels at procedurally required distances evaluated before final selection of label characteristics?

36. What conventions or rules were used to group controls and associated displays? Were these applied consistently?

37. What was the basis for control/display grouping?

38. Is the tone, intensity, periodicity, or location of auditory alarms related in any way to the cause of the alarm or to the position of relevant controls/displays on panels or consoles? What is the relationship?

39. What was the basis for annunciator window groupings?

40. What was the basis for selection of auditory alarms?

41. What conventions or rules were used for the orientation of switch positions (e.g., up=on; down=off)? Were these rules or conventions followed consistently?

42. What was the basis for switch orientation conventions or rules?

43. Was design consideration given to panel operations when the operator is wearing a breathing apparatus and/or protective garments?

44. Was consideration ever given to how much information the operator must be able to correctly recall in order to operate the panel?

45. Was consideration ever given to how much information the operator must process correctly to operate the panel?

46. In selecting panel components, was any consideration given to their maintainability (e.g., replacing light bulbs, changing labels, replacing switches)?

47. Was operator response time required by failures taken into account in the location of various components?

48. What AEC regulations and industry standards were used to guide the panel design?

49. How do you guarantee accessability of redundant Class 1E displays? For the Class 1E controls?

50. Were walk-throughs using mockups, or simulations ever performed to measure or observe operator performance? If so, what measures were taken; what, if any, problems were uncovered; and what changes were made?

51. Were operator performance data collected during plant and control room testing? If yes, what data; what problems were uncovered; and what changes were made?

52. Does the A&E have a program to monitor operator performance or design comments on a continuing basis? If so, what problems have been found, and what backfits made?

53. Was any attempt made to optimize the noise level in the control room? If so, have tests been made periodically to verify calculated (predicted) levels?

54. What basis was used for the acoustics in the control room?

55. Did the A&E participate in developing plant operating procedures? If so, were walk-throughs/simulations used? Did the operators participate? What bases were used for the format, language and organization of the procedures?

56. In what manner and to what degree were operators/maintainer task analyses used to develop and/or evaluate the following:

    1. Operator information and performance requirements
    2. Selection and location of controls and displays
    3. Organization and layout of console panels

57. What was the basis for assigning readouts to panel indicators vs. computer printout?

58. Were control, displays, guards, or other features included on the panel expressly to protect specific (expensive) equipment items from damage? If yes, what features?___"Sync Stick"

APPENDIX

CONTROL ROOM ASSESSMENT

1.0 CONTROL ROOM LAYOUT

1. In your control room, how many physically separate control panels are there? Consider each geometric change as a separate panel.

   No. of panels = ______

2. How are these panels laid out? (Rough Sketch)

3. What functionally different panels are there in your Control Room. Use major functions; such as Coolant Systems, Turbines, Aux. Systems, etc., for your list. Please number your panels on the sketch (question 2) with the appropriate function (1 through 30).

1. ________________	16. ________________
2. ________________	17. ________________
3. ________________	18. ________________
4. ________________	19. ________________
5. ________________	20. ________________
6. ________________	21. ________________
7. ________________	22. ________________
8. ________________	23. ________________
9. ________________	24. ________________
10. ________________	25. ________________
11. ________________	26. ________________
12. ________________	27. ________________
13. ________________	28. ________________
14. ________________	29. ________________
15. ________________	30. ________________

4. Panels are arranged by (check one):

_____ Frequency of Use
_____ Criticality of Systems
_____ Frequency and Criticality
_____ Other Criteria (Specify)

5. Using the list of panels in question above, please circle those panels that make extensive use of mimic or functional control/display grouping.

6. Your panel is designed primarily for (check one):

_____ Seated Operation
_____ Standing Operation
_____ Both

7. Your panels are designed primarily for (check one):

_______ Single Operator Monitoring (normal operation)
_______ Dual Operator Monitoring (normal operation)
_______ Other (explain)

8. When standing in the primary control area of your panel, the operator (check one):

_______ Can read all important displays
_______ Can see all important displays
_______ Must move to another area to see displays

9. Annunciator lights are grouped by system (check one):

_______ Always
_______ Frequently
_______ Sometimes
_______ No

If "no" or "sometimes" use the space below to describe conventions or rules used to group annunciators:

10. Annunciator panels are located above or nearby the controls/displays of the systems they monitor (check one):

_______ Always
_______ Frequently
_______ Sometimes
_______ No

11. How are multiple, simultaneous alarms handled by the operator?

12. Are alarms coded by their severity? (Describe convention)

13. Are chart recordings intended for use by operators under normal or emergency conditions?

2.0 CONTROLS

14. Approximately how many of the following types of Controls are there on your Control Room Panels:

A. Discrete Rotary Control Selector Switch
   - 2 position ___
   - 3 position ___

B. Continuous Rotary Controls
   - Thumbwheels ___
   - Knobs ___
   - Hand Cranks/Wheels ___

C. Push Buttons (Without Legends) ___

D. Legend Switches (Backlighted Pushbuttons) ___

E. Toggle Switches ___

F. "J" Handle Switches
   - 2 position ___
   - 3 position ___

G. Alpha-Numeric Keyboards ___

H. Joysticks or Levers ___

I. Other (Describe) ___

15. What systems are controlled normally by computers?

______ ______

______ ______

16. Does the computer assist the operator in any way other than by reporting status information?

___ No

___ Yes, Explain

3.0 DISPLAYS

17. Approximately how many of the following types of <u>Displays</u> are there on your <u>Control Room Panels</u>:

A. Clock Face Dials:
   Swing Needle Meters ___

B. Strip Chart Recorders ___

C. Digital Counters ___

D. Backlighted Displays (Other than Annunciators) ___

E. Alarm Annunciators ___

F. Single Pointer Gauges:
   Horizontally Oriented ___
   Vertically Oriented ___

G. Double Pointer Gauges:
   Horizontally Oriented ___
   Vertically Oriented ___

H. Single Indicator Light ___

I. Double Indicator Light ___

J. Triple Indicator Light ___

K. Cathode Ray Tube Displays ___

L. Video Displays ___

M. Photographic Displays ___

N. Other Indicator Lights ___

18. Approximately how many auditory signals for alarms or attention devices are there in your control room? (Check as appropriate)

( )	Telephones	___	( )	Bells	___
( )	Radio Com.	___	( )	Buzzers	___
( )	System Alarm Annunciators	___	( )	Tone	___

( )	Emergency	___	( )	Sirens	___
( )	Other	___	( )	Other	___

19. Of the total number of labels for controls and displays in your Control Room, approximately how many have been modified with additional stickers, tabs or notations to provide new or current information to the operators?

    No. = ______

20. Who usually makes up and affixes these stickers or notes?

    ____

21. Please give the meanings (e.g., red = open) for each of the colors listed below. List all meanings for each color.

    White = __________

    Yellow (Amber) = __________

    Red = __________

    Green = __________

    Blue = __________

    Other (Specify) = __________

22. What means are used to display trend data to the operators? (Please list variables displayed.)

Strip Chart	Computer Printout	Video Display	Operator-Prepared Charts	Other

23. Was functional/system outlining and summary nomenclature used?

24. Please outline the change of shift procedure.

25. Would it be possible to get one copy of Reactor Trip and LOCA procedures for review?

26. Where are procedures located?

27. How are they organized?

28. Is there a procedure for translating Operator Comments into backfits or procedure changes?

    ___ Yes, Describe

    ___ No

29. What major backfits have been made since licensing? (List)

30. Describe the communications network serving the operator.

APPENDIX T

LIST OF SELECTED HUMAN ENGINEERING REFERENCES AVAILABLE PRIOR TO 1970

APPENDIX T

LIST OF SELECTED HUMAN ENGINEERING REFERENCES AVAILABLE PRIOR TO 1970

Askren, W.B. (ed.): Symposium on reliability of human performance in work, AMRL, TR 67-88, May, 1967.

Bennett, E., J. Degan, and J. Spiegel (eds.): Human factors in technology, McGraw-Hill Book Company, New York, 1963.

Bilodeau, E.A., and Ina M. Bilodeau (eds.): Principles of skill acquisition, Academic Press, Inc., New York, 1969.

Bradley, J.V., and N.E. Stump: Minimum allowable knob crowding, USAF, WADC, TR 55-455, December, 1955.

Chapanis, A.: "Human engineering," in C.D. Flagle, W.H. Huggins, and R.H. Roy (eds.), Operations research and systems engineering, chap. 19, pp. 534-582, The John Hopkins Press, Baltimore, 1960.

Chapanis, A.: Man-machine engineering, Wadsworth Publishing Company, Inc., Belmont, Calif., 1965.

Damon, A., H.W. Stoudt, and R.A. McFarland: The human body in equipment design, Harvard University Press, Cambridge, Mass., 1966.

Fogel, L.J.: Biotechnology: concepts and applications, Prentice-Hall, Inc., Englewood Cliffs, N.J., 1963.

Folley, J.D., Jr. (ed.): Human factors methods for system design, The American Institute for Research, Pittsburgh, 1960.

Gagne', R.M. (ed.): Psychological principles in system development, Holt, Rinehart and Winston, Inc., New York, 1962.

Handbook of instructions for aerospace personnel subsystem design (HIAPSD), USAF, AFSC Manual, 80-3, 1967.

Harris, D.H., and F.B. Chaney: Human factors in quality assurance, John Wiley & Sons, Inc., New York, 1969.

Irwin, I.A., J.J. Levitz, and A.M. Freed: "Human reliability in the performance of maintenance," in Proceedings, Symposium on quantification of human performance, Aug. 17-19, 1964, Albuquerque, New Mexico, pp. 143-198, M-5.7 Subcommittee on Human Factors, Electronic Industries Association.

McCormick, E.J.: Human Factors Engineering, (2nd Edition), McGraw-Hill, Inc., New York, 1964.

Meister, D., and G.F. Rabideau: Human factors evaluation in system development, John Wiley & Sons, Inc., New York, 1965.

Morgan, C.T., J.S. Cook, III, A. Chapanis, and M.W. Lund (eds.): Human engineering guide to equipment design, McGraw-Hill Book Company, New York, 1963.

Munger, S.J.: An index of electronic equipment operability: evaluation booklet, The American Institute for Research, Pittsburgh, 1962.

Murrell, K.F.H.: Human performance in industry, Reinhold Publishing Corporation, New York, 1965.

Nadler, G.: Work design, Richard D. Irwin, Inc., Homewood, Ill., 1963.

Personnel subsystems, USAF, AFSC design handbook, ser. 1-0, General, AFSC DH 1-3, Jan. 1, 1969, Headquarters, AFSC.

Rigby, L.V.: The Sandia human error rate bank (SHERB), paper presented at Symposium on Man-Machine Effectiveness Analysis: Techniques and Requirements, Human Factors Society, Los Angeles Chapter, Santa Monica, Calif., June 15, 1967.

Sandberg, K.O.W., and H.O. Lipshultz: Maximum limits of working areas on vertical surfaces, USN ONR SDC, Report 166-1-8, reprint, April, 1952.

Stevens, S.S. (ed.): Handbook of experimental psychology, John Wiley & Sons, Inc., New York, 1951.

Van Cott, H.P., and J.W. Altman: Procedures for including human engineering factors in the development of weapon systems, USAF, WADC, TR 56-488, October, 1956.

Webb, P. (ed): Bioastronautics data book, NASA, Washington, D.C., NASA SP-3006, 1964.

Weight, height, and selected body dimensions of adults: United States, 1960-1962. Data from National Health Survey, USPHS Publication 1000, series 11, no. 8, June, 1965.

Woodson, W.E. and D.W. Conover: Human engineering guide for equipment designers, (2d ed.), University of California Press, Berkeley, 1964.

APPENDIX U

COMPARISON OF PLANTS ON DESIGN DEVELOPMENT ISSUES

COMPARISON OF PLANTS ON DESIGN DEVELOPMENT ISSUES

DESIGN DEVELOPMENT ISSUE	TMI-2	CALVERT CLIFFS-1	OCONEE-3
Review of panel design with respect to operation	Not B&R's responsibility (2)	Yes, preliminary procedures from CE ran on mockup	Yes, performed in walk-throughs by plant personnel as there were no formal procedures yet. (3)
Use of Operator opinion during design	Yes. (1)	Yes, early in panel design (5)	Yes, in mockup phase (3)
Selection of alarm and annunciator strategies	Frequency of flashing, white light, size, shape and alarm horns chosen to match the annunciators included in systems sent by the vendor. (1)	Grouped by system, subsystem, component alarms selected according to manufacturer's standard. (5) BG&E placed and combined the alarms and annunciators, and defined the strategy. (4)	Alarms located near controls or annunciators associated with operator response to alarm. Annunciators standard from vendor. Alarm bell for computer selected for differentiation from other annunciator alarms. (3)
Assessment of readability (displays and labels)	Looked at pictures in catalogs, or held display up to see how far away it was readable. Did not duplicate visual environment in tests. Held up sample letter sizes for labels until a readable size was found. (1)	Yes, used mockup to assess readability. (5)	Yes, mockup and lab tests run on equipment to test readability. (3)
Control/display grouping	Controls near associated indicators, grouped by systems on panels, grouped in flow pattern. (1)	Grouped: functionally; centered; bottom to top sequencing; operationally sequenced. (5)	Controls and displays together for a particular function, grouped by frequency of use. (3)
Design for operator wearing breathing apparatus and/or protective garments	No. (1)	No. (5)	No, in mockup saw no reason to change anything as a result of operator wearing breathing apparatus. (3)
Operator recall/information processing requirements	Not considered. (1) Never tested in a time frame. (2)	No formal considerations. (5)	Not specifically addressed, thought consistency and clarity would eliminate need for memory/recall and reduce information processing needs. (3)

COMPARISON OF PLANTS ON DESIGN DEVELOPMENT ISSUES

DESIGN DEVELOPMENT ISSUE	TMI-2	CALVERT CLIFFS-1	OCONEE-3
Maintainability	Obtained samples to ascertain maintainability, looked for "rugged" controls. (1)	Yes, maintainability was considered in BG&E review. (6)	Yes, except in case of systems provided by vendors, lab tests were run on ease of calibration and serviceability. (3)
Operator response times (considered in panel design?)	No. (1&2)	No. (5)	No. (3)
Use of mockups, walk-throughs and simulators	No. (1)	Yes. (5)	Yes. (3)
Noise level (taken into account?)	No. (1)	No. (5)	No, not optimized, but minimized with carpeting. (3)
Participation in developing procedures	Yes, drafted a few (1) initial drafts. (2)	No, but reviewed some. (5)	Yes. (3)
Task analyses (were they performed?)	No. (1&2)	No. (5)	No. (3)
Design to protect expensive equipment	Yes, location (2) and selection (1) of controls and displays	Yes, interlock controls for expensive equipment (5), sync. stick - RC pumps. (4)	No information.

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
NUREG/CR-1270, Vol. III
Appendices, Part 2

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
T.B. Malone and others

5. DATE REPORT COMPLETED

MONTH	YEAR
December	1979

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
The Essex Corporation
333 N. Fairfax St.
Alexandria, Va.

DATE REPORT ISSUED

MONTH	YEAR
January	1980

6. *(Leave blank)*

8. *(Leave blank)*

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Three Mile Island Special Inquiry Group
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.
04-76-293-08

13. TYPE OF REPORT
Final

PERIOD COVERED *(Inclusive dates)*

15. SUPPLEMENTARY NOTES

14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

This report describes a study of human factors engineering aspects of the Three Mile Island-2 (TMI-2) accident on 28 March 1979. The objective of the study was to evaluate the causal contributions, if any, of operator performance and effects on operator performance of:

- Control room design
- Operator training
- Emergency procedures

The topic of the current report is the degree to which operator errors were, in turn, caused by human factors engineering aspects of the control room design, operator training, and emergency procedures.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS *(This report)*
Unclassified

21. NO. OF PAGES

20. SECURITY CLASS *(This page)*
Unclassified

22. PRICE
S

NRC FORM 335 (7-77)

*U.S. GOVERNMENT PRINTING OFFICE : 1980 0-308-886/6767

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

POSTAGE AND FEES PAID
U.S. NUCLEAR REGULATORY
COMMISSION

U.S. MAIL