TOLEDO
EDISON

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Docket No. 50-346

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May 10, 1982

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US NUCLEAR REGULATORY COMMISSION
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Mr. John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stolz:

After completing your initial review of our August 4, 1980 submittal on information concerning post Three Mile Island action items set forth in NUREG 0660 and NUREG 0737 items:

I.a.2.1 Immediate Upgrading of Senior Reactor Operator Training and Qualifications and

II.b.4 Training for Mitigating Core Damage,

additional questions requiring our response were generated.

Attached is the additional information requested on your questions 1 through 6. Also included as an enclosure for Item II.b.4 is an outline of the training program for the Mitigation of Core Damage and other related outlines of programs relating to the training on mitigation of core damage.

Very truly yours,

R P Crouse

RPC/JMH/bjs

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Enclosures:
1. Operator Training - Mitigation of Core Damage
2. Simulator Training Program Documentation
3. Simulator Training for Mitigation of Core Damage

cc: DB-1 NRC Resident Inspector

1. The use of installed instrumentation was addressed in the core damage training by the following methods:

   a. Use of the Self-Powered Neutron Detector System (SPND) to determine that high temperatures exist within the reactor core. This is accomplished by observing the alarm summary status for the SPND System. The limitations of this method were also stressed to the operator in that the thermionic emission currents and space charge release currents are very small and only meant to be an indication that high temperatures occurred and not a means of determining actual core temperature.

   b. The use of the core exit installed thermocouples was discussed in detail. This included normal methods of reading the incore thermocouples via the plant computer and the use of the local readings of the thermocouple taken by the I&C Department.

   c. Methods of interpreting out-of-core source range nuclear instrumentation readings to core uncovery and inadequate core cooling were described. This included a discussion of how coolant temperature, coolant level and coolant void distribution can be related to inadequate core cooling using out-of-core nuclear instruments.

   d. Methods of determining effective core flowrate utilizing various heat balance procedures were provided. This enables the operator to calculate an expected core flow and compare it to installed indications.

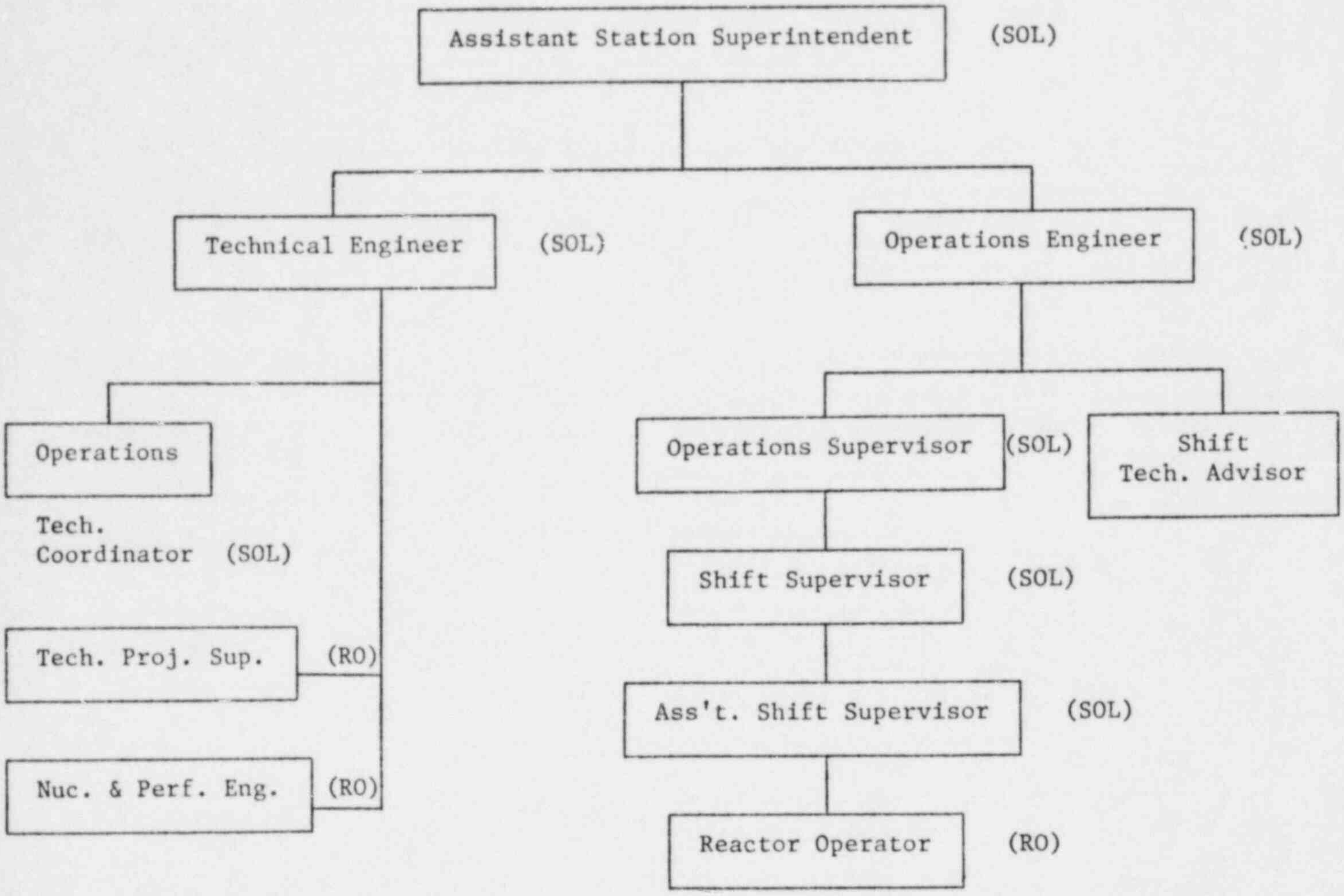
   e. Void size calculations were provided to demonstrate how the operator can determine approximate sizes of steam or non-condensible gas bubbles. This will better indicate the actual coolant volume present in the system and whether the core is actually covered.

   f. Failure modes of instruments were addressed, especially radiological monitoring instruments for the Containment and letdown systems. Included in this discussion were the results of the Sandia National Laboratory study of the TMI Dome Monitor.

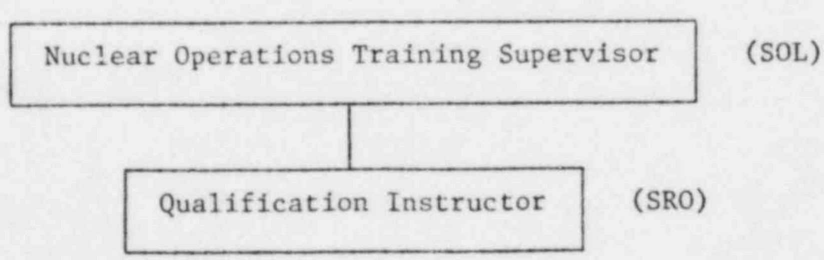
   g. The Davis-Besse Station also ran tests on determining pressurizer level using the change in resistance of the heater banks when uncovery of the bank occurs.

2. To date a quiz specifically addressing accident mitigation has not been administered. The requalification quizzes are given in the subject titles of the NRC examination. Therefore, questions concerning safety systems and procedures are given when these lectures are presented. For example the present requalification lecture series discusses the OTSG tube leak emergency plan along with the actual transients that occurred at Oconee and Ginna Nuclear Power Stations. The quiz on this lecture is required to be taken by all licensed operators and shift technical advisors.

The Operations Manning Chart is as follows:



also



NOTE: SOL = Senior Operator License

RO = Operator License

3. The Davis-Besse Operator Training Program has been significantly changed to include much more emphasis on plant transient behavior. Prior to TMI the licensed operator training program Pressurized Water Reactor Technology (PWR) Program consisted of 160 hours of lectures on plant systems with most of the training on the individual system configurations. The program now consists of 280 hours classroom instruction. The additional time is used to discuss not only the systems but their effects on all aspects of operation normal and abnormal.

   The simulator training for license candidates has also been significantly modified. The first 80 hours (40 hours lecture, 40 hours on the simulator) deals primarily with startup certification and low power plant operations. Following this the student commences part of the Control Room watchstanding under instruction. The student then is given lectures on emergency and abnormal procedures, along with accident diagnosis and mitigation. Following these lectures an additional 80 hours at the simulator is provided where the license candidate maneuvers the plant for power operations, transients, emergencies, and accidents. After simulator training the operator returns to the Control Room for final watch training under instruction.

   The last phase of the operator program (Specialized License Training) consists of an audit examination similar in scope to the actual NRC license examination. Based on the results of this examination, the Specialized License Training lecture series is presented which is approximately 200 hours of advanced PWR, technical specifications, reactor theory, and thermodynamics.

4. The instructors at Davis-Besse who have administered licensed operator training hold a Senior Reactor Operator license. They are cognizant of operating history, problems and procedures by the fact that they initiate the required reading and lectures for the operators. Also, they review all Licensee Event Reports, Transient Assessment Reports, etc. for operation of Davis-Besse and of other Babcock & Wilcox plants. Currently one instructor holds a Reactor Operator license; however, he is in training for the next Senior Reactor Operator license examination in December 1982. This instructor will not be teaching systems or plant responses addressed in the Denton letter. Also, all examinations and lectures given are reviewed and approved by the Nuclear Operations Training Supervisor who does hold a Senior Reactor Operator license. All training instructors are required to take part in all aspects of the requalification program which includes a minimum actual plant watchstanding requirement of 24 hours per quarter.

5. The Davis-Besse Requalification Program does mandate an accelerated requalification for any operator scoring less than 80% overall or any category less than 70%. In addition, the operator is removed from license duties during the accelerated requalification. All operators scoring less than 80% in any category is required by procedure to attend that lecture series for the category below 80%.

6. The Babcock & Wilcox simulator now has a computerized program listing each operator by name and by position at the control panel (ie, RO, Ass't. RO, SS, etc.). The program lists the events performed by name and by reference number to the Denton letter. Several events cannot be performed on the simulator such as "loss of instrument air". For these events the operators are given an in-plant walkthru by the training staff or other designated examiner, and these are entered in appropriate records. The combination of simulator and in-plant walkthrus allow completion of the Denton letter items in the allowed time frame.

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

December 8, 1980

TRAINING INFORMATION NOTICE

SIMULATOR TRAINING PROGRAM DOCUMENTATION

The requirements to perform certain control manipulations are specified in 10CFR Part 55 as part of the operator requalification program. These requirements are further defined in Enclosure 4 of H.R. Denton, NRC, letter to All Power Reactor Applicants and Licensees dated March 28, 1980, and in draft ANS 3.1 Standard on Qualification and Training of Personnel for Nuclear Power Plants. Provisions are made to allow completion of these control manipulations at a simulator facility. The Babcock and Wilcox Simulator Training Center is capable of providing simulation for most of the required control manipulations.

The Babcock and Wilcox Training Center provides documentation to each customer describing what drills and evolutions each individual performed during training at the simulator facility. Effective with training completed the week starting December 8, 1980 a revised format for the reporting of training will be used. A copy of this new format is attached. Each plant training staff will be able to keep track of individual participation in the required evolution and drills by direct correlation of B&W report with the individual's annual training requirements.

The Training Summary Sheet represents our effort to provide documentation of training performed in an improved format that reduces confusion and allows you to readily determine the status of each operator in meeting the required control manipulations for his requalification program. Any questions or comments concerning this method of documentation should be to W. H. Odell, Manager, Instruction Services.

Sincerely,

N. S. Elliott, Manager
Training Services

NSE:hcv

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Encl. 2

SIMULATOR TRAINING SUMMARY SHEET

__________________________ has completed a -week training program consisting of hours of classroom time and hours of simulator operations. The time spent in the simulator consisted of performing the evolutions listed below in the indicated capacity with the remainder of the time devoted to manual and automatic ICS power operations.

SS = Shift Supervisor
F = Foreman
RO = Reactor Operator
ARO = Assistant Reactor Operator
STA = Shift Technical Advisor

Evolutions Performed	SS	F	RO	ARO	STA		
(1) Plant or reactor startups to include a range that reactivity feedback from nuclear heat addition is noticeable and heatup rate is established.							
(a) Reactor Startup (to 10^{-8} Amps)							
(b) Reactor Starutp (to 5% Power)							
(c) Reactor Startup (to 15% Power)							
(d) Reactor Startup (to 100% Power)							
(e) Plant Temp Change Due to Heatup/ Cooldown (^{o}F)							
(f) Calculate a ECP							
(g) Plot a 1/m							

(2) Plant Shutdown.

(a) Power Reduction (100% to 15% Power)							
(b) Power Reduction (15% to Hot Shutdown)							

Evolutions Performed	SS	F	RO	ARO	STA		
(3) Manual Control of Steam Generators and/or feedwater during startup and shutdown.							
(a) Main and Startup Valves in Manual During Startup							
(b) Main Feedwater Pumps in Manual During Startup							
(4) Boration and or Dilution During Power Operation.							
(a) Adjust Rod Height for Rod Index Curve Limits							
(5) Any significant (> 10%) Power Changes in Manual Rod Control.							
(a) Defeat Neutron Error Signal from ICS							
(b) Diamond in Manual							
(6) Any Reactor Power Change of 10% or Greater Where Load is Performed with Load Limit Control or Where Flux, Temperature, or Speed Control is on Manual.							
(a) Reactor Power Level Change > 10% Power With Either Main Turbine, Rod Demand, Diamond, Main Feedwater Valves and/or Main Feedwater Pumps in Manual.							

Evolutions Performed	SS	F	RO	ARO	STA
(7) Loss of Coolant.					
(a) Significant O.T.S.G. Tube Leak a capacity (#/sec)					
(b) Reactor Coolant Leak Inside Primary Containment (Capacity #/sec)					
(c) Reactor Coolant Leak Outside Primary Containment (Capacity #/sec)					
(d) Leak-Rate Determination					
(e) Saturated Reactor Coolant Response (PWR)					

(8) Loss of Instrument Air (If Simulated Plant Specific)

Not Simulated.

(9) Loss of Electrical Power (and/or Degraded Power Sources).					
(a) Blackout					
(b) Fail 6900 Vac Bus					
(c) Fail Startup Transformer					
(d) Fail Diesel Generator					

(10) Loss of Core Coolant Flow/Natural Circulation.					
(a) Loss of Power to All Reactor Coolant Pumps					
(b) Actuation of Safeguards System Which Require Shutting Off RCP's due to Low Pressure Condition (Leak, Overcooling, etc.)					

Evolutions Performed	SS	F	RO	ARO	STA		
(11) Loss of Condenser Vacuum.							
(a) Failure of Condenser Vacuum Breaker (Open/Closed)							
(b) Fail Air Ejector							
(c) Fail Condenser 3-Way Valve							
(d) Fail Gland Seal Steam Supply Valve							
(e) Fail Condenser CW Pumps							
(12) Loss of Service Water if Required for Safety.							
NOTE: The Canal Water System Cools Components Associated with Service Water.							
(a) Fail Canal Water Pumps							
(b) Fail SU Transformer #1							
(13) Loss of Shutdown Cooling.							
a) Failure of Decay Heat Pump During S/D Cooling							
(b) Fail NSCW Valve to DH Cooler							
(c) Fail Open Decay Heat Bypass Valve							
(d) Fail NSCW Pumps							

Evolutions Performed	SS	F	RO	ARO	STA		
(14) Loss of Component Cooling System or Cooling to an Individual Component.							
(a) Failure of Component Cooling and Containment Penetration Valve (Open/Shut)							
(b) Fail Component Cooling Water Pumps							
(c) Fail CCW Valve to Letdown Cooler							
(15) Loss of Normal Feedwater or Normal Feedwater System Failure.							
(a) Failure of Condensate Pump(s) which Cause Loss of Main Feedwater Pump(s)							
(b) Loss of Main Feedwater Pump(s)							
(c) Fail Main Feedwater Control Valve							
(d) FAil Main Feedwater Block Valve							
(e) Fail Startup Feedwater Control Valve							
(f) Fail Startup Block Valve							
(g) Vary Feed Pump Speed Feedback Signal							
(h) Degrade Feed Pump Speed							
(16) Loss of All Feedwater (Normal & Emergency).							
(a) Fail Emergency and Main Feedwater Pumps							
(b) Fail Emergency and Main Feedwater Block Valves							
(c) Fail Emergency and Main Feedwater Control Valves							

Evolutions Performed

(17) Loss of Protective System Channel.

(a) Defeat RPS Trip Functions							
(b) Defeat ESFAS Automatic Initiation							

(18) Mispositioned Control Rod or Rods (or Rod Drops).

(a) Dropped Rod							
(b) Stuck Rod							

(19) Inability to Drive Control Rods.

(a) Motor Fault							
(b) Retard Rod Motion							

(20) Conditions Requiring Use of Emergency Boration.

Note: Emergency Boration Will Occur Whenever the HPI Pumps Take a Suction From the BWST.

Evolutions Performed	SS	F	RO	ARO	STA		
(21) Fuel Cladding Failure or High Activity in Reactor Coolant or Offgas.							
(a) Failed Fuel Casualty							
(22) Turbine or Generator Trip							
(a) Load Rejection							
(b) Turbine Trip							
(c) Defeat ICS Signal to EHC							
NOTE: Record for All Casualties When Result is a Reactor Trip							
(23) Malfunction or Automatic Control System(s) Which Affect Reactivity.							
Note: These Malfunctions Can be Satisfied Under Sections (18) or (19).							
(24) Malfunction of reactor coolant pressure/volume control system.							
(a) Loss of Makeup Pump(s)							
(b) Failure of Pressurizer Spray Valve (Open/Shut)							
(c) Failure of Letdown Isolation Valve (Open/Shut)							
(d) Failure of Makeup Valve (Open/Shut)							
(e) Failure of PORV							
(f) Failure of Pressurizer Relief Valve							

Evolutions Performed	SS	F	RO	ARO	STA		
(24) Continued							
(g) Failure of 3-Way Valve							
(h) Fail Pressurizer Level Control Valve							

(25) Reactor Trip.

NOTE: Record for all Casualties Which Result in a Reactor Trip.

(26) Main Steam Line Break.

(a) Inside Containment (Capability ______ #/sec)							
(b) Outside Containment (Capacity ______ #/sec)							

(27) Nuclear Instrumentation Failure(s).

(a) Failure of Power Range NI							
(b) Failure of CIC							
(c) Failure of P.C.							

Evolutions Performed	SS	F	RO	ARO	STA		
(28) Instrument Failures.							
(a) Fail Selected TH Instrument High/Low							
(b) Fail Selected TC Instrument High/Low							
(c) Fail RCS Flow Signal to ICS							
(d) Fail Header Pressure Signal to ICS							
(e) Fail Feedwater Flow Signal to ICS							
(f) Fail OTSG Startup Level to ICS							
(g) Fail OTSG Operate Level to ICS							
(h) Fail Neutron Error							
(i) Fail Steam Generator Pressure Signal							
(j) Fail Feedwater Temperature Signal							
(k) Fail Power Range Signal to ICS							
(l) Fail Feed Pump ΔP							
(m) Fail Generated Megawatts							
(n) Fail Tave							
(o) Fail RCS Pressure Signal							
(p) Fail Pressurizer Level Signal							
(29) Unassigned Casualties and Evolutions.							
(a) RC Pump Trip							
(b) Plant Temperature Changes $\geq 50^{\circ}F$							
(c) TMI Demonstration							
(d) Solid Plant Operations							
(e) Draw a Bubble in the Pressurizer							
(f) Fail Seal Water Control Valve							
(g) Fail NSRW Pump							
(h) Degrade Low Pressure Feed Heater							
(i) Degrade High Pressure Feed Heater							
(j) Fail Plant Cooling Water Pump							
(k) Fail Turbine Plant Cooling Water Pump							
(l) Fail RB Emergency Coolers							
(m) Fail RB Spray Pumps							

Evolutions Performed	SS	F	RO	ARO	STA		
(29) Continued							
(n) Fail HP Injection Valves							
(o) Fail Boron Addition Valve							
(p) Turbine Trip Locked Out on Reactor Trip							
(q) Fail Turbine Bypass Valve							
(r) Degrade Secondary Steam Relief Setpoint							
(s) Fail Heater Drain Pump							

ADDITIONAL PAGE FOR (29) UNASSIGNED CASUALTIES AND EVOLUTIONS

Evolutions Performed	SS	F	RO	ARO	STA	

TECO 1981 REQUALIFICATION

SIMULATOR

MONDAY

1. Reactor start-up safety rods withdrawn to 100% Power

   Fail CIC during S/U

   Fail F. W. during S/U

   Feedwater pump in manual during S/U

   Manual control of reactor, feedwater and turbine during power increase (document manual operations)

TUESDAY

1. Tube leak 500 gpm (also document plant shutdown and cooldown)
2. Small break in RC System outside Reactor Building (letdown or makeup system)
   a. Determine leak rate
   b. Locate and isolate
3. Loss of Power to all RC pumps

   Natural circulation cooldown to 500^{o}

   (document natural circulation)

WEDNESDAY

1. Loss of component cooling water to letdown coolers
2. Loss of condenser vacuum (start with small steam leak then cause steam reducer valve to fail losing auxiliary steam to air ejectors)
3. Loss of all feedwater normal and emergency (document solid plant operation)
4. Loss of RPS
5. Reactor trip, turbine fails to trip
6. ICS failures (temperature, levels, feedwater temperature, etc.)

Encl. 3

THURSDAY

Adjust rod heights for rod index curve by changing boron concentration (make calculation necessary to support exercise)

- Loss of cooling accident - leak large enough to make the RC reach saturation conditions and not repressurize (document emergency boration after actuation of HPI.) (document operation at saturated conditions) carry out to long term cooling (suction of LPI from CV sump)
- Reactor trip with secondary safety valve stuck open

FRIDAY

Blackout with both diesels failed (feedwater available)

15-20 min repair one diesel

Return normal power 35-40 min.

REQUALIFICATION TRAINING

FOR

TECO 1981

CLASS ROOM SCHEDULE

CONTROL ROOM SCHEDULE

Day No.	Day/Date	Time	Subject	Reference Instructor	Time	Operation	Reference Instructor
		07:30 to 09:30	REVIEW DECAY HEAT & METHODS TO REMOVE DECAY HEAT NATURAL CIRCULATION RC SYSTEM AT SAT DURING COOLDOWN	TRAINING SERVICES			
		09:30 to 11:30	GAS/STEAM BINDING AFFECTS ON CORE COOLING BORON PRECIPITATION CONCERS FOLLOWING LOCA	ED ANDERSON			
		07:30 to 10:30	CONSEQUENCES OF INADEQUATE CORE COOLING AND AND LIKELY CORE DAMAGE EFFECTS	TOM THORNTON			
		10:30 to 11:30	USE OF SPND'S IN RECOGNITION OF DEGRADED CORE CONDITIONS	TRAINING SERVICES			
		07:30 to 09:30	DETECTION AND TREATMENT OF INADEQUATE CORE COOLING USING CORE EXIT THERMOCOUPLES	TRAINING SERVICES			
		09:30 to 10:30	THERMOCOUPLES AND CORE FLOW BLOCKAGE RELATED TO TMI-2	TRAINING SERVICES			
		10:30 to 11:30	RELATIONSHIP OF OUT OF CORE SOURCE RANGE TO DEGRADED CORE CONDITION	TRAINING SERVICES			
		07:30 to 11:30	RELEASE OF FISSION PRODUCTS & FISSION PRODUCT TRANSPORT RESPONSE OF GAMMA RADIATION MONITORS CHEMICAL AND RADIOCHEMICAL SAMPLING PROBLEMS	DON NITTI			
		07:30 to 09:30	REVIEW OF OPERATING WITH CORE DAMAGE	TRAINING SERVICES			
		09:30 to 11:30	REVIEW OF RELATED TAP'S THAT COULD EFFECT TECO	TRAINING SERVICES			

TECO

Name

BABCOCK & WILCOX
NUCLEAR TRAINING CENTER
LYNCHBURG, VIRGINIA