TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

INSPECTION

R-80-04-SQN

Subject: Tennessee Valley Authority

Sequoyah Nuclear Plant

(1) Operator Training and Qualification Program and
(2) Emergency and Abnormal Operating Instructions

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I. Scope

The inspection included a review of the Reactor Operator (RO)/Senior Reactor Operator (SRO) Training and Requalification Training Programs; verification of the implementation of these programs; and verification of the adequacy of qualification of reactor operators, senior reactor operators, operations supervision and operations advisory personnel. The scope of the inspection also included a review of the Emergency Operating Instructions (EOI's) and the Abnormal Operating Instructions (AOI's) to determine the conceptual adequacy of these procedures.

II. Conclusions

A. RO and SRO Training and Requalification Training Programs and Implementation

1. The initial training and requalification programs meet the NRC requirements except that the plant administrative controls do not ensure that reactor operators or senior operators will be cognizant of changes to the facility, changes to procedures, and changes to the license.

2. Neither plant nor division administrative controls ensured that training program material content will be upgraded as required on a timely basis.

3. Division Procedure Manual DPM No. N79A12, "Review of Training Required by Division Procedures," had not specified the frequency of the reviews.

4. Neither the Division Procedures Manual nor the SQN Administrative Instructions specified the evaluation and examination grading criteria for the RO/SRO initial training programs.

5. The Training Center was developing a program to certify RO/SRO training of instructors in the techniques of instruction; however, plant management stated that they did not intend to send senior operators who were participating as part-time instructors through the program.

6. The third week of the RO/SRO requalification program had not been formalized.

B. Qualification of Operations and Support Personnel

1. The RO's and SRO's assigned to SQN-1 are considered qualified to operate SQN based on the following:

a. Successful completion of the TVA RO and SRO Training and Requalification Programs.

b. Successful completion of the NRC licensing examination and receipt of RO and SRO NRC licenses.

c. Experience received by SQN RO's and SRO's during preoperational testing, hot functional testing, low power testing, simulator manipulations, and observation and familiarization training at other operating PWR's.

2. The RO's and SRO's lack actual operating experience at an operating PWR. NSRS does not perceive this to be a potential safety problem during power escalation and normal operation because the operators had received supplementary and directly applicable training and experience as follows:

   a. Academic training on reactor theory and operation of SQN systems.

   b. On-the-job training for operating SQN systems.

   c. Intensive simulator training on normal, abnormal, and emergency reactor operation.

   d. Experience gained during preoperational and hot functional testing.

   e. Actual operating experience to be gained during low power testing.

   f. Observation training at other operating PWR's.

   g. Operating experience gained operating small research reactors.

C. Emergency Operating Instructions

As described in NUREG-0660 Draft 12/10/79, task I.C, section B, step 4.1, NRC is engaging in a long-term effort to upgrade operating procedures, with the goal being to develop a relatively small number of symptom-oriented procedures which will consider operator error, single and multiple active failures, and passive failures. The Report of the President's Commission on the Accident at Three Mile Island (Kemeny Report), recommendation D.4.a, and the "TVA Nuclear Program Review: Sequoyah Nuclear Plant and the Report of the President's Commission on the Accident at Three Mile Island," section D.4.a, similarly point to enlargement of the operating procedures to consider multiple failures and human errors. Therefore, the conclusions of this report concerning the adequacy of the EOI's and AOI's apply to the revision in effect at the time the review was performed. NSRS does not contend that the EOI's and AOI's should remain fixed, but rather they should evolve as additional studies are made.

The Emergency Operating Instructions were conceptually adequate except for the following:

1. Two important emergencies were not covered--the recovery from degraded core conditions and the loss of all AC power.

2. The lists of symptoms in some EOI's omitted important indicators.

3. Incorrect referrences for Technical Specification limits were found.

4. Incorrect automatic actions were described.

5. Caution notes were used where action steps would have been appropriate.

6. The operator was referred to other EOI's in the immediate operator action steps rather than having the necessary steps presented directly.

7. Where normal versus off-normal values of plant parameters were decision factors, numeric values for normal conditions were sometimes missing.

D. Abnormal Operating Instructions

The Abnormal Operating Instructions were conceptually adequate except for the following:

1. No procedure was provided for "loss of protective system channel" as required by Regulatory Guide 1.33.

2. Conclusions 2 through 7 of item C above apply also to the AOI's.

III. Recommendations

A. RO and SRO Training and Requalification Training Program and Implementation

1. Review the administrative controls and revise to ensure that licensed reactor operators and senior reactor operators are made cognizant of changes to the facility, changes to procedures, and changes to the license in conformance with 10CFR55, Appendix A, and TVA's Operational Quality Assurance Manual.

2. Provide a formal means to ensure that the training program material has been approved by management and is current.

3. Revise DPM No. N79A12, "Review of Training Required by Division Procedures," to require a review and evaluation of the training program by division personnel each two years.

4. Revise DPM's and AI-14 to include appropriate evaluation/ examination grading criteria for the RO/SRO initial training programs.

5. Certify all RO/SRO instructors in the techniques of instruction. (Include SRO's who are participating as part-time instructors.)

6. Formalize the third week of the Requalification Program.

B. Qualification of Operations and Support Personnel

Provide Westinghouse consultant personnel during testing periods during power ascension and initial power operations.

C. Emergency Operating Instructions and Abnormal Operating Instructions

1. The category I comments in Appendix A on EOI's and AOI's should be evaluated, resolved with NSRS, and incorporated into the instructions prior to power escalation above the five percent level.

2. The category II comments in Appendix A on EOI's and AOI's should be evaluated and incorporated into the instructions as deemed appropriate by the Division of Nuclear Power.

3. The Division of Nuclear Power should perform appropriate reactor operator and senior operator training on the revised EOI's and AOI's prior to power escalation above the five percent power level.

IV. Details

A. TVA's RO and SRO Training Program

The NRC's requirements for operator training are stated in 10CFR55. To meet these requirements, TVA had developed basic criteria which were presented in the FSAR, section 13.2, and in N-OQAM, part III, section 6.1, items 1.5 and 1.5.2. The following documents implement the training program described in these documents: Administrative Instruction AI-14, section II.D; DPM No. N78A13, sections I, II, III.A, III.C, III.D, III.F, III.G, III.H, and III.I; DPM No. N75A5; and DPM No. N7704.

Since 1971 TVA primarily had three groups of RO/SRO training groups. Group 1 candidates were trained by Westinghouse, and Groups 2 and 3 were trained by TVA. Groups 2 and 3 essentially received the same training except that Group 3 received the majority of the classroom training at the Power Production Training Center while Group 2 received the classroom training at SQN.

Over the years TVA's operator training program had evolved into what is now known as the Nuclear Student Generating Plant Operator (NSGPO) Training Program. However, prior to this candidates for RO and/or SRO licenses at SQN received varying degrees of training, depending upon their past operating experience. The programs that the various SQN training groups participated in are described as follows:

1. Initial RO/SRO Training Programs

   Prior to the NSGPO program, the initial candidates consisting of Group 1 personnel participated in a different program that was conducted by Westinghouse and TVA. This training consisted of the following:

   a. Fossil Student Generating Plant Operator Training Program - This was a 24-month program conducted by TVA designed to train personnel with no previous steam plant experience and covered power plant basics; print reading; plant auxiliary equipment; instrumentation; steam turbochargers; electrical training; boilers; and on-the-job training. Progress throughout the program was monitored by prescheduled oral and written examinations.

   b. Basic Nuclear Course - This program involved 11 weeks of basic training in the principles involved in the design, construction, and operation of the reactor.

   c. Onsite Lecture Series - This program involved introducing the operator to plant equipment and systems. This course consisted of 16 weeks of lectures and on-the-job training at the plant on how individual systems and controls operate.

   d. Offsite Observation Training - This program involved 10 weeks of observation of actual operational activities at an operating PWR. Two weeks were spent familiarizing the candidate with plant systems with emphasis on design requirements and and as-built system parameters. The next eight weeks involved plant familiarization and walkthroughs.

e. Simulator Training - This program involved nine weeks of PWR simulator training at the Westinghouse Nuclear Training Center at Zion, Illinois, simulating normal, abnormal, and emergency plant operation and transients.

f. Reactor Operatings Training - This program involved approximately two weeks of training on a small experimental or testing reactor. Each trainee operated the reactor through at least ten startups. The course also included approach to critical experiments; health physics procedures; waste disposal; control rod calibrations; ion chamber calibrations; function of neutron absorber; xenon experiments; and radioactive material handling under water.

g. Additional Training - The courses listed above were essentially the minimum required to prepare for the NRC examination. The Group 1 candiates participated in additional training which was promulgated due to delays in the SQN construction schedule. This training involved classroom, simulator, and on-the-job training which was basically equivalent to the original training received.

2. Summary of Current RO/SRO Training Program

SQN training Groups 2 and 3 participated in the following training programs. Some of the training was conducted at SQN and some at the Power Production Training Center.

a. Nuclear Student Generating Plant Operator (NSGPO) Training Program - The NSGPO training program was a comprehensive program that prepared a candidate with no previous nuclear experience to enter the reactor operator cold license program.

To enter the NSGPO training program, an applicant was required to:

(1) Be at least 18 years old.

(2) Have a high school diploma.

(3) Have taken the General Aptitude Test Battery examination.

Once an applicant was accepted, he entered a four-phase training program which took 22 months to complete. The first three phases of the program (i.e., 16 months) were taught at the Power Production Training Center. The course work consisted of basic nuclear physics, mechanical and electrical print reading, operating mechanical and electrical print reading, operating instructions, and PWR systems. During the fourth phase, the candidate was transferred to the nuclear plant site to gain additional system knowledge and "hands-on" experience. Upon successful completion of this phase, the candidate was eligible to become an assistant unit operator (AUO). Successful completion of each phase not only entailed completion of all course requirements but also required passing both an oral and a written examination. The oral examination required the candidate to demonstrate a knowledge of operating procedures and was administered and graded by a subcommittee. The written examination required the candidate to demonstrate a knowledge of the technical aspects of the related subjects. This examination was also graded and administered by a local subcommittee.

b. Cold License Training - Those candidates who had completed the NSGPO training program were eligible to become AUO's. After being an AUO for five months, the Training Review Board, which consisted of three persons from Operations Plant Management, reviewed the qualifications of the AUO and determined whether the individual was a suitable candidate for cold license training. The cold license training program was conducted at the Power Production Training Center and consisted of four weeks of classroom lectures on systems and procedures, seven weeks of simulator manipulation on normal and abnormal reactor operations combined with classroom lectures, one week of review, and a final week of oral and written examinations. Candidates who successfully passed this course were ready for NRC examinations.

c. Offsite Observation Training - This training consisted of eight weeks of observation training at an operating PWR.

d. Reactor Operations Training - This training involved approximately one week of training at a small reactor (either at Oak Ridge National Laboratory or at Georgia Tech Training Reactor).

e. Additional Training - The courses listed above were essentially the minimum required under TVA's program; however, Groups 2 and 3 participated in additional training subsequent to the initial training to ensure that their knowledge and skills of SQN remained at a high level.

3. General RO/SRO Training

RO's and SRO's were required to participate in a variety of training. These programs involved training on industrial safety; fire protection and prevention; health physics; plant security; emergency plans; quality assurance; quality control; plant administrative instructions; document control; work requests; clearance procedures; adverse conditions and corrective action; plant surveillance programs; temporary conditions; plant modifications; procurement and material control; special processes and tests; and fire bridgade member refresher training. In addition, SRO's in shift engineer or assistant shift engineer positions received training on cleanliness criteria for piping systems; fire bridgade leadership; site Radiological Emergency Plan director; and electrical systems.

4. NSRS Review

NSRS reviewed the RO/SRO training program and the associated administrative controls; reviewed qualification records; and conducted discussions with reactor operators, senior operators, operations supervision, instructors, and training center supervision to determine the adequacy of the program and to verify implementation of the program. The following observations are offered.

a. The Division of Nuclear Power had developed administrative procedures which assigned responsibilities and provided criteria for the establishment and implementation of an RO and SRO training program that was in conformance with the NRC regulations and was being implemented.

b. The administrative controls did not provide a formal means for upgrading the training program to ensure that training program content was current. The program was being upgraded as changes were recognized by operations and training center management, instructors, and training coordinators; and it appeared to be current. The program had been reviewed to assure that lessons learned from TMI were incorporated into the program.

c. DPM No. N79A12, "Operational Review of Training Required by Division Procedures," dated November 23, 1979, provided for a review by division personnel of the training program to ensure adequate instruction, course material, and the proper attitude of both instructor and trainees. The procedure did not, however, establish requirements for the frequency at which the review would be accomplished. Discussions with division personnel indicated that DPM No. N79A12 was being personnel indicated that DPM No. N79A12 was being revised to include a frequency of review. Discussions with division personnel also indicated that the initial review was in progress and that it involved attendance of training, review of examination/ evaluation results, and review of program content.

NSRS did not evaluate the effectiveness of this program because it had only recently been initiated; however, the scope of the review appeared adequate.

d. Discussions with unit operators (three of four interviewed) indicated they felt that the training program would be strengthened by extending the minimum time requirements for an auxiliary unit operator to be eligible for cold license training to enable the assistant unit operators more on-the-job training in the plant. TVA's Nuclear Program Review (Blue Book) commits TVA to increasing the time-in-grade requirements for assistant unit operators by six months. Discussions with the plant training coordinator indicated that the upgrading of this program was in progress and would pertain to operators in the future.

e. The Blue Book also commits TVA to:

(1) increase the NSGPO Training Program to 26 months,

(2) expand requalifications training for licensed reactor operators and reactor operator candidates to include a unique simulator training device for each type of TVA reactor,

(3) increase operator salaries,

(4) establish basic intelligence tests for all operators and students being selected into the training programs, and

(5) pursue a long-range goal of having the operator training program accredited as a program culminating in a recognized academic degree or certification.

NSRS verified that the Division of Nuclear Power had completed or was working on these objectives as they apply to SQN. Items (1) and (2) had been completed; and items (3), (4), and (5) were in various stages of development.

f. Subsequent to the Three Mile Island (TMI) accident, Essex Corporation prepared a report for the NRC's Special Inquiry Group entitled "Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2," NUREG/CR-1270. The findings of Essex were that the TMI-2 training program was in full compliance with government imposed standards concerning training; however, certain deficiencies in the training program were noted as follows:

(1) Only 5 percent of training time is used for simulator training.

(2) Training in emergency procedures was deficient.

(3) Training failed to provide for measurement of operator capabilities.

(4) Training of instructors was deficient.

(5) Training was not closely associated with procedures.

(6) Training generally ignored the fact that operators were dealing with a slowly responding system.

(7) The training program at TMI-2 did not provide formal upgrading of methods, materials, and course content.

During review of the training program, NSRS has concluded that the identified deficiencies do not exist in TVA's training program with two exceptions [items (4) and (7)] which are discussed in paragraphs IV.A.4.b and IV.A.4.g of this report.

g. The training center was developing a program to certify the simulator instructors on various instructional techniques. SQN furnished instructors to participate in the third week of the Requalification Program. Discussions with plant management indicate that they had no intention of requiring instructors to participate in the instructor certification program.

h. On March 29, 1980, the NRC issued a letter to all Power Reactor Applicants and Licensees pertaining to revised criteria to be used by the NRC staff in evaluating reactor operator training and licensing that can be implemented under the current regulations and to establish an effective date for their implementation.

NSRS has reviewed the revised NRC criteria and has compared them to the requirements of the SQN RO and SRO Training and Requalification Program. NSRS comments are contained in Appendix B to this report.

1. TVA's cold license and requalification training programs involved simulator sessions for seven and two weeks, respectively. The sessions consisted of classroom instruction in the normal, emergency, and abnormal operating procedures followed by simulation of various accidents and events which required knowledge of the various operating procedures. Initially simulator training progressed from normal system startups's with operator cognizance of the tasks to be performed to the point where emergency conditions with multiple failures were programed into the simulator without operator cognizance. Operators were evaluated on their response. When simulating emergency conditions, SQN simulator instructors could choose any of 21 different initial plant conditions based on power level and fuel burnup. Additionally, one or more malfunctions from a list of approximately 140 events could be simulated by the instructor. The simulator had the capability to process 12 malfunctions simultaneously, each with varying degrees of severity. As the sequence of events proceeded, the instructor, if he desired, could take a "snapshot" of the conditions existing at any time during the event (i.e., the computer could store these conditions for later recall). In addition, the instructor could backtrack the simulator (up to four hours) to any point in the event for instruction purposes.

   The instructor had the option to allow the event to proceed in real time, slow time (i.e., eight times slower than real time) or fast time (i.e., 32 times faster than real time). The simulator also had the capability of monitoring 200 parameters but only 50 at any particular time. Additionally, the simulator had the capability of providing an historical record of 50 parameters (every ten seconds), the instructor's actions, and the operator's actions to aid the instructor and the operator in correcting operating problems that may have been encountered during the sequence of events.

   In conclusion, NSRS believes that the SQN simulator training program is closely associated with the SQN normal, abnormal, and emergency operating procedures and that the normal, abnormal, and emergency conditions simulated will adequately prepare the operator to operate the plant safely.

j. The RO/SRO training programs produced 23 persons with NRC SRO licenses and 11 with NRC RO licenses for a total of 34. Of the 34 persons that received licenses, 16 were trained by Westinghoues cold license programs, and 18 were trained by TVA. Of the Westinghouse trained personnel who took NRC licensing examinations, two failed a portion of the examination, but passed a reexamination, and two failed and were dropped out of positions requiring licenses. Of the TVA trained personnel who took the NRC examinations, three personnel originally failed a portion of the examination--one passed a reexamination, one unit operator who had passed the RO examination but failed the SRO examination elected not to retake the SRO examination, and one passed an RO examination but failed the SRO examination and had not yet been reexamined. In addition, two personnel trained by TVA were licensed by the NRC at Browns Ferry. Both of these personnel failed their SRO NRC examinations for SQN. One of these persons had been assigned duties that did not require a license, and one passed an RO examination.

In conclusion, it should be noted that although it may not be readily evident from the foregoing presentation, a higher percentage of personnel in the later training groups were passing NRC examinations on the first attempt. Of the nine personnel that did not pass portions of the NRC examinations, only one was in the last group of 12 applicants. The training for this group was primarily performed at the Power Production Training Center, and it becomes more significant by the fact that the new NRC grading criteria was applied to this group.

k. TVA's training program for all phases of SQN RO/SRO training required examinations or evaluation of the RO/SRO by the instructors or supervisors to ensure that the candidate was demonstrating adequate skills and knowledge at intermediate and final stages of training. Typically, the grading criteria was not discussed in the administrative controls (except for the requalification training program which is discussed in paragraph IV.B of this report); however, in practice the same grading criteria that was applied to the requalification program was being applied to the cold license programs.

NSRS reviewed several of the cold license program examinations, evaluation results and examination results and concluded that the Division of Nuclear

Power was implementing an adequate program for measurement of operator effectiveness. However, evaluation and grading criteria should be specified in the administrative controls.

l. The Office of Power Nuclear Safety Review Board (NSRB) had performed a review of TVA's training programs more than a year ago. According to discussions with division personnel, the NSRB had not been involved in specific reviews of SQN RO/SRO qualifications and training. However, NRC requirements for auditing the performance, training, and qualifications of the entire unit staff, annually, were not effective until issuance of the SQN operation license on February 29, 1980.

m. Discussions with division personnel indicated that a task group had been assigned to upgrade the Operational Quality Assurance Manual (OQAM) and corresponding division procedures to ensure that the procedures were compatible with the requirements in the OQAM. No date was given for the completion of this project.

B. <u>RO and SRO Requalification Training Program and Program Implementation</u>

1. <u>TVA's Requalification Program</u>

The NRC had set forth the requalification program for licensed operators (Reactor Operators and Senior Reactor Operators) in 10CFR55, Appendix A. TVA satisfied these requirements by developing a program described in FSAR section 13.2 and in OQAM part III, section 6.1, paragraph 1.5. These requirements were implemented at the division level by DPM No. N78A13, section IIIB, and at the plant level by Administrative Procedure AI-14.

Basically, the operators received 120 hours of retraining at least every two years. Eighty hours of the retraining consisted of a lecture series for both RO's and SRO's. The Power Production Training Center instructors and the Sequoyah Nuclear Plant staff covered the following topics:

a. Theory and Principles of Operation
b. General and Specific Plant Operating Characteristics
c. Plant Instrumentation and Control Systems
d. Plant Protection Systems
e. Engineered Safeguard Systems
f. Normal, Abnormal, and Emergency Operating Procedures
g. Radiation Control and Safety

h. Technical Specifications
i. Applicable Portions of NRC Rules and Regulations
j. Spill Prevention Control
k. Site Radiological Emergency Plan Director Training
l. Administrative Instructions

During 40 hours of the retraining program, each operator manipulated the Sequoyah simulator located at the Power Production Training Center. Each operator performed at least ten different reactivity control manipulations. In addition, the operators performed certain functions utilizing normal, abnormal, and emergency operating instructions.

The RO and SRO retraining differed at this point in that the SRO retraining emphasized leadership and development of the SRO's ability to oversee operation of the plant, especially during an accident situation.

During the classroom lectures, written quizzes were given. A grade of less than 80 percent required a repeat of lecture attendance and the quiz.

Upon completion of retraining, each licensed operator was given a written examination which required approximately six to eight hours to complete. SRO's and RO's were given different examinations, and the SRO's were expected to go into greater depth on certain questions. When an operator received a score of less than 70 percent in any category of the annual written examination, the Training Review Board reviewed the operator's records and determined retraining requirements and the duration of the retraining. Removal from licensed duties and placement in an accelerated retraining program was mandatory when an overall score of less than 80 percent on the annual written examination was received. Upon completion of the accelerated retraining, the operator must score at least 80 percent overall on the written examination which covered the area or areas of his indicated weakness. After successfully completing the examination and approval from the Training Review Board, the operator could resume his licensed duties.

Each operator received two evaluations. The shift engineer evaluated each operator's performance on shift, and the training instructor evaluated each operator during the simulation of normal, abnormal, and emergency conditions.

The operations supervisor maintained the records for each operator. The records included copies of written examinations and the operator's answers, results of evaluations, documentation of additional training, and documentation of at least control manipulations.

In addition, the program required that each operator be supplied periodically with facility design changes, procedure changes, and license changes. Each operator was required to read and document that he has read the procedure. Moreover, before an operator could begin his shift, he was required to read and document that he had read any AOI or EOI that had been revised since his last shift. The retraining program was conducted by the Sequoyah plant staff and the Training Center instructors.

2. NSRS Review

NSRS reviewed the Requalification Program and the associated administrative controls; conducted discussions with RO's and SRO's; and reviewed operator records to determine the adequacy of the program and to verify implementation of the program. The following observations are offered:

a. The Division of Nuclear Power had developed administrative procedures which assigned responsibilities and provided criteria for the establishment and implementation of an RO and SRO retraining program in conformance with the NRC regulations except as described below:

   10CFR55, Appendix A, and N-OQAM, part III, section 6.1, require that the operator be cognizant of changes to the facility, changes to procedures, and changes to the license. N-OQAM, part III, section 6.1, additionally, stated that records would be maintained that demonstrate this cognizance.

   A review of the plant administrative controls indicate that there was no provision to ensure that these requirements would be satisfied. AI-5, "Shift Turnover," required the operator to document review of changes to the EOI's and AOI's prior to assuming responsibility for the shift. Operations Section Letter OSLA 76, September 6, 1979, "Shift Group Training," provided for weekly training sessions which were conducted on SQN Licensee Event Reports; appropriate Clearing House releases, significant changes to Standard Practices and Operation Section letters, changes to Technical Specifications, recent equipment problems, operator errors, and general plant status. A review of meeting records and discussions with the training coordinator indicated that significant design changes and procedure changes were also discussed. The fact remains that all changes were not discussed and there had not been a requirement for all RO's/SRO's to attend these training sessions. The training sessions were held each week with the shift in training status. Each shift rotated into training status every fifth week.

Significant changes may be discussed during the yearly operator requalification sessions; however, NSRS is concerned with the fact that the administrative mechanisms do not provide for complete, timely dissemination to the operators of design changes, changes to the license, or changes to the procedures.

b. The third week of the Requalification Training Program had not been formalized; however, a draft program had been developed. Plant management stated that the detailed plan for the third Requalification week would be formalized.

c. The Division of Nuclear Power had implemented the Requalification Training Program in 1979 for the SQN RO's and SRO's, and it was continuing at the time of the inspection. The final examinations had not been given; however, TVA had committed to the NRC to apply the latest grading criteria (70 percent each category and 80 percent overall) to the examinations. All RO's and SRO's were participating in the requalification program.

d. Paragraphs IV.A.4.b, .c, .g, and .i also apply to the requalification training program.

e. The Power Production Training Center maintained controlled copies of all procedures and the Technical Specifications. This ensured that training on the simulator was conducted with the latest revision of the procedures. Design change packages were reviewed by SQN operations management and only sent to the Power Production Training Center if it involved training or involved a change to the simulator. The Power Production Training Center did maintain a controlled set of SQN drawings, however, which reflected the design changes.

C. <u>Assessment of the Qualificatons of Operations and Support Personnel</u>

1. <u>RO's and SRO's</u>

The TVA operators and senior operators meet or exceed the qualification requirements endorsed by the NRC at the time of the inspection. All reactor operators/senior operators in operating positions which require NRC licenses had been examined by the NRC and had received their license.

TVA senior operators and supervisors typically have had extensive fossil power plant operating experience and had been the recipient of extensive academic, simulator, and on-the-job training prior to and during preoperational and hot functional testing. Training had continued even though delays in loading fuel had been encountered. The original training group or RO's/SRO's (all but two are now senior operators) have had approximately twice the amount of training required by the NRC to obtain a license. All RO's/SRO's have had yearly simulator practice subsequent to their primary training. SQN had implemented the retraining program during 1979 and were applying the latest NRC examination grading criteria to the program (80 percent overall and at least 70 percent in each category of the examination).

The basic skill of the operator is derived from having complete familiarity with the specific systems and their operating range and being able to diagnose abnormal and emergency operating conditions and respond accordingly. An operator acquires this skill by experience in operating a reactor that is essentially identical to the one for which he is being licensed and by the use of a simulator that is representative of the reactor for which he is being licensed. Due to TVA's policy of acquiring operators up through the ranks, and since SQN is TVA's first PWR, the plant operators lack actual operating experience on an operating PWR. However, they have had extensive experience on a simulator specifically designed and programed to simulate SQN controls and response. SQN operators had also acquired valuable experience during the performance of preoperational and hot functional testing. It should be noted that actual PWR operating experience is not required by the NRC to obtain and maintain an NRC license. Additional experience at another PWR that is not identical to SQN or at a reactor that has not experienced abnormal operation may not provide experience as valuable as simulator experience. The necessary experience can be obtained by performing control activities on a simulator of the type (Westinghouse PWR) for the license being sought. Experience prior to fuel loading had been obtained by SQN RO's and SRO's by performing manipulations on small reactors, by performing observation training at operating PWR's, by operating a simulator, and by operating SQN during preoperational and hot functional testing.

Additional experience will be gained during the low power testing. A portion of this testing involves performance of a series of natural recirculation tests which are being accomplished by SQN at the request of the NRC for the first time. These tests will be repeated such that each shift will gain the experience of performing the test.

2. Support Personnel (Consultants)

To compensate for the lack of actual PWR operating experience, TVA had provided the following for the duration of low power testing in addition to the normal complement of personnel required by the Technical Specifications.

a. Consultant personnel from Nuclear Startup Services (NSS) Corporation who previously held NRC licenses at operating PWR's to act as advisors to the Shift Engineers.

b. Consultant startup engineers from Westinghouse to provide advice to the Shift Engineers.

c. Operations management consultant with extensive operating experience.

Items a and b above were necessary to satisfy the license conditions to ensure that sufficient personnel with PWR experience would be available during low power testing.

NSRS reviewed the resumes of the consultants, except those of the Westinghouse startup engineers which were not available, and concluded that these personnel met the qualifications established by the NRC in the Technical Specifications; however, they had not all been trained in Westinghouse plants and had not been trained at SQN. A discussion with the leader of the NSS Corporation consultants at the site indicated that the consultants on shift had undertaken self study to become cognizant of the SQN systems, procedures, and the Technical Specifications. Discussions with SQN shift engineers indicated that they felt these consultants would be useful in providing advice during conduct of surveillance testing; however, SQN management was skeptical about the basis for the requirement and the practical necessity for their presence. A certain degree of skepticism is understandable in view of the fact that TVA does have a strong operator training program that has provided well qualified operators when compared with previous requirements for operator qualifications. Since TVA has not operated a PWR before, the training program provided the best qualified operators possible short of licensing them at a PWR being operated by another utility.

NSRS believes that the consultants would be more useful if their experience had been in the operation of Westinghouse ice condenser type plants or if they had undergone a formal training program at the SQN simulator; however, NSRS would not advocate the removal of these personnel during low power testing since their presence could represent some additional degree of safety margin.

NSRS does believe that the presence of the Westinghouse consultants who presumably will be experts in PWR startups would enhance the safety margin during testing during power escalation and initial full power operations.

D. Review of Emergency Operating Instructions (EOI's) and Abnormal Operating Instructions (AOI's)

A review of the Emergency and Abnormal Operating Instructions currently in use at SQN was conducted. The NSRS scope of review was conceptual in nature, and the following general guidelines were used to verify instruction validity:

1. Consistent with applicable Regulatory Guides.
2. Consistent with commitments specified in the FSAR and Technical Specifications.
3. Clear diagnostic instructions for identifying the particular condition.
4. Clarity in terms of individual operator actions and cautions.
5. Effectiveness with which instructions could be carried out.
6. Assurance that the necessary operator actions could be performed.

Appendix A contains the comments generated as a result of the NSRS review.

V. Personnel Contacted

A. Sequoyah Nuclear Plant

*J. M. Ballentine, Plant Superintendent
*J. R. Bynum
**D. J. Record, Operations Supervisor
***G. G. Wilson, Assistant Operations Supervisor
***J. R. Walker, Training Coordinator
*R. L. Hamilton, QA Supervisor
**C. T. Benton, Shift Engineer
**W. R. Ramsey, Shift Engineer

**R. E. Yarborough, Shift Engineer
**W. O. Lovelace, Assistant Shift Engineer
**L. E. Guinn, Unit Operator
**L. M. Hodges, Unit Operator
**W. G. Payne, Unit Operator
**H. A. Tirey, Unit Operator

B. Division of Nuclear Power

J. Olson, Nuclear Operations Staff
F. Szczepanski, Chairman, Nuclear Safety Review Board

C. Power Production Training Center

**R. J. Johnson, Coordinator
**J. Mantooth, Assistant Coordinator
**C. T. Brewer, Instructor
**S. W. McNair, Instructor
**C. H. Noe, Instructor

D. Nuclear Startup Services Corporation

**F. E. Flynn, Consultant
**W. P. Johnson, Consultant

E. Watts Bar Nuclear Plant

*G. Denton, Operations Supervisor

F. Nuclear Regulatory Commission

*W. T. Cottle, Resident Inspector

*Present at the exit interview on March 27, 1980.
**Personnel interviewed.
***Personnel interviewed; also present at exit interview on March 27, 1980.

VI. Documents Reviewed

10CFR55, "Operators' Licenses," 1979.

Sequoyah Nuclear Plant Final Safety Analysis Report, Section 13.2.

Sequoyah Nuclear Plant Unit 1, Technical Specifications, February 29, 1980.

Letter from D. F. Ross to H. G. Parris dated February 29, 1980, entitled, "Sequoyah Nuclear Plant, Unit 1 - Issuance of License No. DPR-77."

Sequoyah Nuclear Plant Operational Quality Assurance Manual, Part II, Section 4.2A, and Part III, Section 6.1, January 23, 1979.

Administrative Instruction AI-4, "Plant Instructions - Document Control," Revision 25, February 14, 1980.

Administrative Instruction AI-2, "Authorities and Responsibilities for Safe Operation and Shutdown," Revision 9, March 1, 1980.

Administrative Instruction AI-5, "Shift Relief and Turnover," Revision 5, February 1, 1980.

Administrative Instruction AI-6, "Log Entries and Review," Revision 5, August 23, 1979.

Administrative Instruction AI-14, "Plant Training Program," Revision 7, February 29, 1980.

ANSI N18.1 - 1971, "Selection and Training of Nuclear Power Plant Personnel," March 8, 1971.

ANS 3.1 - 1978, "American National Standard for Selection and Training of Nuclear Power Plant Personnel."

NUREG-0011, Supplement No. 1 to the Safety Evaluation Report by the Office of NRR USNRC in the Matter of TVA Sequoyah Nuclear Plant, Units 1 and 2, Docket No. 50-327 and 50-328, February 22, 1980.

NUREG/CR-1270, "Three Mile Island - Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island - 2"

Sequoyah Safety Evaluation Report, Part II, TMI-2 Issues Related to Fuel Load and Low Power Test Program.

TVA Nuclear Program Review, May 1979.

TVA Nuclear Program Review: Sequoyah Nuclear Plant and the Report of the President's Commission on the Accident at Three Mile Island, November 1979.

US NRC Regulatory Guide 1.8, "Personnel Selection and Training," Revision 1, September 1975.

TVA, P PROD, "Nuclear Generating Plant Operator Training Programs," DPM No. N78A13.

TVA, P PROD, "Turnover of Shift Responsibilities," DPM No. N74A16.
TVA, P PROD, "Nuclear Steam Generating Plant Operator (NSGPO) Training Program," DPM No. N75A5.

TVA, P PROD, "Power Production Training Center Operating Information and Modifications," DPM No. N75A9.

TVA, P PROD, "Operational Review of Training Required by Divisions Procedures," DPM No. N79A12.

TVA, P PROD, "Assistant Unit Operator Training," DPM No. N7704.

TVA, P PROD, "Nuclear Plant Licensed Operations Shift Personnel Responsibilities," DPM No. N7903.

TVA, P PROD, "Shift and Relief Turnover," DPM No. N7904.

TVA, P PROD, "Nuclear Plant Licensed Operations Shift Management Responsibilities," DPM No. N7905.

Sequoyah Nuclear Plant, Operations Section, "Abnormal and Emergency Instruction Revisions Review Acknowledgement," OSLA81, February 7, 1980.

Sequoyah Nuclear Plant, Operations Section, "Training - 4th Period Student Operators," March 2, 1978.

TVA, Power Operations Training Center, Pressurized Water Reactor, "Cold License Certification Program."

Pressurized Water Reactor Systems Manual, Volumes 1 and 2.

Sequoyah Nuclear Plant Requalification Training Program Weeks 1 and 2, 1979, Prepared by: C. O. Brewer, S. W. McNair, and C. H. Noe.

Sequoyah Nuclear Plant, Operations Section, "Training Standard," OSLA50, November 6, 1978.

Introduction to Nuclear Power, TVA, Second Edition, Seventh Printing, November 1979.

American National Standard Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, ANSI N18.7-1976.

Resumes of NSS Personnel Qualifications

Personnel Qualification Records

EOI-4 Revision 3, Emergency Boration

EOI-5 Revision 5, Station Blackout

EOI-6 Revision 5, Loss of Reactor Coolant Flow

EOI-7 Revision 5, Control Room Inaccessibility

EOI-8 Revision 2, Fuel Cladding Failure or High Activity in Reactor Coolant

EOI-9 Revision 5, Dropped or Damaged Fuel Assembly

EOI-10 Revision 3, Plant Fires

EOI-11 Revision 3, Abnormal Release of Radioactive Materials

EOI-12 Revision 3, Emergency Shutdown

EOI-13 Revision 3, Chlorine Release

EOI-14 Revision 1, Anticipated Transient Without Scram

AOI-1 Revision 6, Reactor Trip

AOI-2 Revision 4, Malfunction of Reactor Control System

AOI-3 Revision 3, Malfunction of Reactor Makeup Cotnrol

AOI-4 Revision 4, Nuclear Instrumentation Malfunctions

AOI-5 Revision 3, Operation with One Reactor Coolant Loop Out of Service

AOI-6 Revision 6, Excess Primary Plant Leakage

AOI-7 Revision 2, Probable Maximum Flood

AOI-8 Revision 3, Tornado Watch

AOI-9 Revision 2, Earthquake

AOI-10 Revision 3, Loss of Control Air

AOI-11 Revision 1, Loss of Condenser Vacuum

AOI-12 Revision 1, Loss of Containment Integrity

AOI-13 Revision 2, Loss of Essential Raw Cooling Water

AOI-14 Revision 3, Loss of RHR Shutdown Cooling

AOI-15 Revision 3, Loss of Component Cooling Water

AOI-16 Revision 3, Loss of Normal Feedwater

AOI-17 Revision 4, Turbine and Generator Trips

AOI-18 Revision 2, Malfunction of Pressurizer Pressure Control System

AOI-19 Revision 4, Inadvertent Safety Injection

AOI-20 Revision 2, Malfunction of Pressurizer Level Control System

AOI-21.1 Revision 3, Loss of 125V DC Vital Battery Board I (Unit 1)

AOI-21.2 Revision 3, Loss of 125V DC Vital Battery Board II (Unit 1)

AOI-21.3 Revision 3, Loss of 125V DC Vital Battery Board III (Unit 1)

AOI-21.4 Revision 3, Loss of 125V DC Vital Battery Board IV (Unit 1)

AOI-21.5 Revision 1, Loss of 125V DC Vital Battery Board I (Unit 2)

AOI-21.6 Revision 1, Loss of 125V DC Vital Battery Board II (Unit 2)

AOI-21.7 Revision 1, Loss of 125V DC Vital Battery Board III (Unit 2)

AOI-21.8 Revision 1, Loss of 125V DC Vital Battery Board IV (Unit 2)

AOI-22 Revision 1, Break of Downstream Dam

AOI-23 Revision 1, Reactor Coolant Pump Seal Abnormalities

AOI-24 Revision 0, Steam Generator Tube Leak

EOI-0, Revision 0, Immediate Actions and Diagnostics

EOI-1, Revision 10, Loss of Reactor Coolant

EOI-2, Revision 9, Loss of Secondary Coolant

EOI-3, Revision 7, Steam Generator Tube Rupture

APPENDIX A

REVIEW OF SEQUOYAH NUCLEAR PLANT EMERGENCY OPERATING INSTRUCTIONS AND ABNORMAL OPERATING INSTRUCTIONS

A review of the Emergency and Abnormal Operating Instructions currently in use at Sequoyah Nuclear Plant was conducted from March 18-April 16, 1980. The comments that resulted are presented in three sections. The first section contains general comments pertaining to several instructions or to the entire set. The second section is devoted to Emergency Operating Instructions (EOI's), and the third section is devoted to Abnormal Operating Instructions (AOI's).

The comments are alphabetically designated to aid further discussion and are categorized to show NSRS judgment of their priority. Category I comments will ensure that (1) the instructions provide technically correct actions for the operator to perform and (2) incorrect and misleading information which has a direct and immediate effect on operator actions are eliminated from the instructions.

Category II comments are also included. NSRS feels that the instructions are adequate without immediate implementation of Category II comments. This is based on the fact that the comments are made to (1) further clarify the instructions and (2) provide additional information. Examples include additional symptoms, updated references, improved format, and additional explanation of equipment operation.

1. <u>Comments Applicable to both EOI's and AOI's</u>

   a. <u>Category I</u>

      (1) Some instructions omit the automatic action that should have occurred during an event and the operator action to verify and manually initiate the automatic action if it did not occur.

   b. <u>Category II</u>

      (1) Typically, the instructions did not include specific numeric values at which action must be taken; did not provide a consistent degree of operator aid in locating specific instruments and controls by number, panel, etc.; and had caution notes which should have been action steps.

      (2) Instructions frequently referred to a normal state or an abnormal value of a parameter. Numerical values for those parameters in the normal state were generally not given, so that the operator would have an increased risk of misjudging what is normal.

2. Comments on Emergency Operating Instructions

a. Emergency Operating Instruction Program

(1) Category II

(a) A procedure for recovery from degraded core conditions had not been developed in accordance with the memorandum from J. R. Calhoun to L. M. Mills dated February 5, 1980 (L33 800205 804) and stipulation 2.C.4.K of the Sequoyah license. The Sequoyah Safety Evaluation Report part II, section I.C.I, discussed the need for a procedure to address degraded core conditions.

b. EOI-0, Revision 0, Immediate Actions and Diagnostics

(1) Category I

(a) Section III.B - This section directed the operator to stop all reactor coolant pumps when pressure decreases to 1300 psig. Westinghouse and EN DES had not reviewed the analysis to determine if this was the correct value. Their input should be considered on this (see NUREG-0623).

(b) Section III.G.6 - This section indicated 40° F subcooling as a requirement for resetting safety injection. This value should be verified by Westinghouse and EN DES to take into account instrument errors and to provide the necessary margin.

(c) The diagnostic portion of the instruction did not provide operator action for reactor coolant line break outside containment. An example is a postulated failure of the letdown line. Although the primary system will respond as described in the instruction, the containment will not respond as the operator is led to believe.

(2) Category II

(a) Section I, Note - The instruction stated that redundant channels should be checked for consistency. The instruction failed to provide operator guidance in the situation where inconsistency exists. The discussion should also include effects of instrumentation failures due to the initiating event [i.e., pipe break severing an instrument line(s)]. The postulated pipe break study is a good reference for this information. The section also failed to address the single failure effects on symptoms and subsequent conflicting information available to the operator.

(b) Sections II.B.4 and II.E.3 - The instruction did not provide a list of equipment actuated by containment isolation or SI actuation. The instruction should include a list of equipment isolated by containment phase A, equipment isolated by containment phase B, and equipment actuated by SI actuation to enable the operator to verify proper actuation.

(c) Sections II.B.4 and II.E.3 - Containment phase B isolation and SI actuation can deceive the operator relative to his interpretation of symptoms given in section I.A and would result in his inability to operate certain equipment--for example:

(aa) Air will be lost to the pressurizer power operated relief valves on phase B isolation and cannot be reestablished to the valves from the main control room.

(bb) If phase B isolation is not reset, the containment spray pump hand switch must be put in "pull to lock" to stop the running pump(s).

(cc) The charging flow is isolated upon SI actuation; and, therefore, for a large LOCA the charging flow will not increase.

(dd) Steam generator blowdown is isolated on SI actuation; and, therefore, the radiation cannot be measured following a LOCA.

Inclusion of equipment lists as discussed in comment 2.b.(1)(b) would aid the operator's diagnostics of accident symptoms.

(d) Section II.B.5 - The instruction did not caution the operator to assure that the steam generators are not overfilled and subsequently filling the steam lines. The steam lines are not designed to withstand the dead weight of water in the line.

(e) Sections II.B.7 and II.B.8 - The instruction did not caution the operator to communicate with the non-accident unit operator regarding common systems and his actions which could adversely affect system(s) operability on the accident unit, for example, assuring throttling back on non-accident unit's component cooling water during the loss of a power train.

(f) Section Purpose - The Instruction stated that in the subsequent documents in this series (EOI-1, -2, and -3), instructions for recovery from the events are presented for the listed accidents. Instructions for spurious actuation of safety injection (one of the listed accidents) is not in this series but is provided in AOI-19.

c. EOI-1, Revision 10, Loss of Reactor Coolant

(1) Category I

(a) Even small leaks of primary coolant and secondary coolant will result in some ice melt. Sequoyah's ice condenser does not contain a lower compartment spray; hence, leakages above that that can be handled by the lower compartment coolers will result in ice melt and the need for the upper compartment spray to condense divider deck bypass leakage. After some ice melt and accumulation of water on the containment floor, the lower compartment coolers will flood and become inoperable. Thus, each small leak will result in significant ice melt and the operation of the upper containment spray.

The ice bed has a finite capacity and some time after an accident will melt through and pass significant energy to the upper compartment. When the melt-through occurs there will be a change in temperature throughout the containment and a rise in containment pressure. In most accidents the containment pressure change will be significant, say, several psi over a period of a few minutes. There is no good way to predict ahead of time when ice melt and the pressure will occur. However, during the accident one can get some indication by the amount of ice melt. The amount of ice melt can be determined from the sump level, the accumulator discharge, and the amount of refueling water and the secondary system water dumped into the containment.

The emergency procedures should contain the information and methodology necessary to approximate the amount of ice melt and roughly product when melt-through occurs.

(b) Westinghouse has not been able to show that their reactor vessel will not fail from flooding of the reactor cavity and quenching the reactor vessel when the primary system is at operating pressure and temperature. Generally speaking, water is kept from the reactor cavity until it is several feet deep in the lower compartment. However, there will be some types of pipe ruptures that could result in water entering in the reactor vessel cavity with primary system pressure and temperature still near operating conditions. This information needs to be provided in the EOI's as a caution to assure that the operator reduces primary system pressure and temperature before the water level reaches the lower vessel head.

(c) Throughout the instruction the terms "station blackout" and "loss of offsite power" are used synonymously. Station blackout implies complete loss of offsite and onsite AC power. The instruction was only applicable to a loss of offsite power, and the appropriate term "loss of offsite power" should be used.

(d) ECCS Switchover Sequence Sections - Evaluation of the switchover sequence must be accomplished with regard to information provided by Westinghouse in letter No. TVA-7493 dated August 15, 1979.

(2) <u>Category II</u>

(a) Section II.AA, Caution - The instruction discussed the possible cavitation associated with train B residual heat removal but failed to discuss any special operator actions required. Examples of special actions would include:

(aa) Observe system flow limitations

(bb) Observe system pressure limitations

(cc) Maintain continuous observation of pump parameters

(b) Appendix A - Saturation condition of the primary system could be more readily diagnosed by the operator if a graph instead of the current table were used. The graph would use reactor coolant system temperature as the abscissa and reactor coolant system pressure as the ordinate. Three curves indicating saturation, 20° F subcooling, and 50° F subcooling would be utilized.

d. <u>EOI-2, Revision 9, Loss of Secondary Coolant</u>

(1) <u>Category I</u>

(a) Section III.F.3.c - Section should read ". . . greater than 2,000 psig, and is . . . ."

(b) Table E.2.2, I.A - Evaluation of the switchover sequence must be accomplished with regard to information provided by Westinghouse in letter No. TVA-7493 dated August 15, 1979.

(2) Category II

(a) Section III, Note - The instruction stated that redundant channels should be checked for consistency. The instruction failed to provide operator guidance in the situation where inconsistency exists. The discussion should also include effects of instrumentation failures due to the initiating event [i.e., pipe break severing an instrument line(s)]. The postulated pipe break study is a good reference for this information. The section also failed to address the single failure effects on symptoms and subsequent conflicting information available to the operator.

e. EOI-3, Revision 7, Steam Generator Tube Rupture

(1) Category I

(a) Section II.K - Throughout the instruction the terms "station blackout" and "loss of offsite power" are used synonymously. Station blackout implies complete loss of offsite and onsite AC power. The instruction was only applicable to a loss of offsite power, and the appropriate term "loss of offsite power" should be used.

(2) Category II

(a) Table 1 - Saturation condition of the primary system could be more readily diagnosed by the operator if a graph instead of the current table were used. The graph would use reactor coolant system temperature as the abscissa and reactor coolant system pressure as the ordinate. Three curves indicating saturation, 20° F subcooling, and 50° F subcooling would be utilized.

f. EOI-4, Revision 3, Emergency Boration

(1) Category I

(a) Sections 4A and 4D - The instruction was not clear as to when one portion of the instruction was to be used as opposed to the other. Distinction for use of section 4A or 4D should be specified. The meaning of the term "extreme emergency" used in section 4D was not defined.

(b) Section 4D, II.B - Since the flow of two boric acid transfer pumps normally equal one centrifugal charging pump, this section should read, "Shift the operating boric acid transfer pump to fast speed and start the second transfer pump, fast speed."

(c) Section 4D.II.C - Since the flow of two boric acid transfer pumps normally equal one centrifugal charging pump, this section should read, "Start one centrifugal charging pump A-A or B-B."

(2) Category II

(a) Section 4A, III.C - One indication to monitor boric acid flow was provided. Another means to verify this flow would be appropriate.

(b) Section 4A, III.C - Useful operator information would be provided if flow requirement range in gal/min was specified.

g. EOI-5, Revision 5, Station Blackout

(1) Category I

(a) Station blackout implies complete loss of offsite and onsite AC power. This instruction was only applicable to a loss of offsite power. The appropriate title should be "Loss of Offsite Power."

(b) A procedure had not been developed delineating the operator actions for loss of AC power. NRC's position stated in Supplement No. 1 to the Sequoyah Safety Evaluation Report, Appendix C, paragraph A-44, prescribes such an instruction.

(c) Section III.A - This section directed the operator to follow immediate operator actions of AOI-1. The immediate operator actions should not normally refer the operator to another procedure.

(d) Section IV.A.1 - The statement "Borate to cold shutdown conditions if necessary" was out of context here. It should be relocated to the proper step in the procedure. The instruction did not state when it is needed or how it is done.

(2) Category II

(a) Section I.C - The electrical system is designed so that a 5-second time delay at zero volts is utilized.

(b) Section I.D. - Same as comment (a).

(c) Section I.E. - Same as comment (a).

(d) Section I.F. - Same as comment (a).

(e) Section I.G. - Same as comment (a).

(f) Section IV.B.4, Footnote 1 - The time for the diesel to start and be connected after a loss of offsite power is 11.5 seconds.

(g) Section IV.G.7 - This step appeared to assume an onsite fault was the cause of the loss of offsite power.

(h) Section V.I - The electrical system is designed such that the 6.9-kV unit board transfer is by paralleling the shutdown board with the unit board.

(i) Section VI - The discussion implied that the essential equipment is not tripped. Any time a transfer is made to the diesel generators, the boards are stripped except for a few 480-V loads.

(j) Section VI - The time for the diesel to start and to be connected after a loss of offsite power is 11.5 seconds. This section should read, ". . . shutdown board in approximately 11.5 seconds, the . . . ."

(k) Section V.Y - If cooldown is desired, General Operating Instruction (GOI) 3 should be followed.

(l) Section V.Y - It is not clear if "electrical systems" means the TVA grid system.

h. EOI-6, Revision 5, Loss of Reactor Coolant Flow

(1) Category I

(a) Section IV.A.1 - The acoustic monitoring system is a valid means by which to monitor safety valve position.

i. EOI-7, Revision 5, Control Room Inaccessibility

(1) Category I

(a) Equipment not provided with capability to transfer control to auxiliary mode could result in erratic operation--for example, conflicting instrumentation response and spurious operation of equipment (i.e., pumps, valves, dampers). In addition, the instruction did not identify what instruments have the capability to transfer control to auxiliary mode.

(b) Section 7A, IV - This section directed the unit 2 control room assistant unit operator (AUO) to proceed to the diesel generator building. The unit 2 AUO is not on assigned watch duty at the present time.

(2) Category II

(a) Checklist - Pressurizer heater group C is a backup group, and group D is the control group.

(b) Sections 7A and 7B, I - Other things than fire and smoke can make the main control room inaccessible--chlorine, airborne radioactivity, refrigerant freon, etc.

(c) Sections 7A and 7B, IV.A.1 - There are precautions associated with operations of the diesel generator at non-load or light-load conditions at full speed for extended periods. Reference memorandum from L. M. Mills to J. R. Calhoun dated February 20, 1980, on "Sequoyah Nuclear Plant Units 1 and 2 - Diesel Engine Driven Emergency Generator Operation at Light/No Load."

j. EOI-8, Revision 2, Fuel Cladding Failure or High Activity in Reactor Coolant

(1) Category I

(a) Section 8A, IV.H - The applicable Technical Specification section was not referenced.

(b) Section 8A, V - Final Safety Analysis Report (FSAR) section 16.3.4.8 is not valid.

(c) Section 8B, IV.L - The applicable Technical Specification section was not referenced.

(d) Section 8B, VII - FSAR section 16.3.4.8 is not valid.

(2) Category II

(a) Section 8A, IV - Loose parts monitoring system and thermocouples are valid indications of locating cause.

k. EOI-9, Revision 5, Dropped or Damaged Fuel Assembly

(1) Category I

(a) Section 9B, I - High activity on RM-90-106B, -112B, -130, and -131 (and -100 if transfer valve is open) are valid symptoms.

(b) Section 9B, IV.A - Utilize radiation monitor (0-RM-90-101) and shield building vent monitor (RM-90-100) to determine if gas release has occurred.

l. EOI-10, Revision 3, Plant Fires

(1) Category I

(a) Sections II.A and II.B - A receipt of a fire alarm from the detection system must have operator's immediate attention. These two steps should be reversed.

(b) Section III.A - Conflicting instrumentation response and spurious operation of equipment (i.e., pumps, valves, dampers) may occur.

(c) Section VI and Attachment A - Residual Heat Removal (RHR) pump room coolers were singled out for detailed instructions. Additional systems and equipment necessary to cool down and maintain the plant in a shutdown condition did not have detailed instructions provided.

(d) Section VI and Attachment A - Sections should address unit 1 and 2 residual heat removal pump room coolers since EOI-10 was applicable to units 1 and 2.

(e) Attachment A, Section I - The instruction stated that Attachment A shall be initiated within 2 hours following a loss of residual heat removal pump room(s) coolers. The design basis for 2 hours was not stated. The critical parameter(s) to monitor was not given.

(2) Category II

(a) Section I.A.11 - Trouble alarms, CO_2 storage low level alarms, and suppression system initiation alarms are also received on system 13 pyrotronics console. "Fire alarm on system 13 pyrotronics control room console" is correct terminology for this section.

(b) Section II.B - Correct terminology is, "With a fire present, initiate plant fire alarm and follow section II.D."

(c) Sections I.A.1, I.A.3, and I.A.4 - Correct terminology is "water spray," not "sprinkler."

(d) Section I.A.6 - Correct terminology is "service building sprinkler systems and water spray initiated."

(e) Section I.A.8 - Makeup water may also be alarmed on high temperature.

(f) Section I.A.9 - Yard CO_2, vault-CO_2 storage pressure abnormal may also alarm.

(g) Section I.A.10 - Correct terminology is "water spray," not "sprinkler."

(h) Section II.D - Correct terminology is "area fire suppression systems."

m. <u>EOI-11, Revision 3, Abnormal Release of Radioactive Material</u>

(1) <u>Category I</u>

(a) Section I.C - The listing of section I.C did not provide a one-to-one correlation with sections I.A and I.B.

(b) Section VI - The action levels for gas and liquid conditions should be used for information only. Guidance for responsibilities and actions should be taken directly from the Radiological Emergency Plan.

(2) <u>Category II</u>

(a) Section I.C - It was not apparent that this section is different from section I.B.

n. <u>EOI-12, Revision 3, Emergency Shutdown</u>

No comment.

o. <u>EOI-13, Revision 3, Chlorine Release</u>

(1) <u>Category I</u>

(a) Regulatory Guide 1.95, position C.6., provides action to be included in an instruction for chlorine release. Instruction did not include all actions.

(b) Section III.A - To prevent lower floors of the control building from being exposed to chlorine and having it leak into the MCR due to positive pressure, section should read, ". . . control room isolation, control building, and pressurizing fans . . ."

(c) Section IV.B - This section was not an immediate operator action item. If MCR isolation has been initiated, operators must don air breathing apparatus immediately.

(d) Section VII - Instruction did not define concentration of chlorine when personnel must use self-contained breathing apparatus.

p. <u>EOI-14, Revision 1, Anticipated Transients Without Scram</u>

(1) <u>Category I</u>

(a) Section II.A.1 - Subsections a, b, and c were written to be completed sequentially. The three steps should provide for concurrent action to implement the steps.

(b) Same comment as d. for steps II.B.1.a.1 and II.B.1.a.2.

(c) Same comments as d. for steps II.B.4.a.1(aa) and II.B.4.a.1(bb).

(2) <u>Category II</u>

(a) The instruction needs to be evaluated in conjunction with the latest Westinghouse ATWS analysis for a SQN-type plant to work out the overall logistics of what actions need to be performed when and what time frame the operator has to accomplish various actions (i.e., turbine trip, centrifugal charging pump start, etc.).

Examples: Westinghouse has a high priority to trip turbine immediately in analysis; instruction had this way down on list.

The operation of auxiliary feedwater system (AFWS) during an ATWS is still being debated by Westinghouse. The latest conclusion is that AFWS operation hurts more than it helps.

(b) With peak pressures that will be reached, charging system operation is doubtful. Therefore, operator will not be able to adjust boron concentration.

(c) II.A.1.d - Steps need to be added to remove the power supply from the rods in a timely manner. This may involve deenergizing both the A and B 480-volt unit boards, assuring the rods go in, and then reenergizing the boards. This concept of identifying methods, time frames, and consequences of executing the deenergization maneuvers should be expanded as far as practical (i.e., short of dropping the grid).

(d) II.A.1.d - NUC PR needs to pursue with Westinghouse what to do in case a mechanical rather than electrical common-mode failure prevents a scram from occurring. In this instances the emphasis is on preventing over-cooling of the reactor coolant system before enough boron can be injected to make and maintain the reactor subcritical. A determination of the critical point for initiating corrective measures should be made.

3. Comments on Abnormal Operating Instructions

a. Abnormal Operating Instruction Program

(1) Category I

(a) A procedure has not been developed for "loss of protective system channel" as specified in Regulatory Guide 1.33.

b. AOI-1, Revision 6, Reactor Trip

(1) Category II

(a) Section IV.A - The operator was referred to EOI-1, -2, or -3, all based on the same symptom, safety injection actuation. It would speed the operator's response if AOI-1 guided him to a specific EOI.

(b) Section IV.A - The term "power blackout" is misleading when only offsite power loss is intended. See related comment for EOI-5.

(c) Section IV.B.1, Caution - The acoustic monitoring system is a valid means by which to monitor safety valve position.

c. AOI-2, Revision 4, Malfunction of Reactor Control System

(1) Category II

(a) Section 2B, Item IV - The parameters noted in 2A.IV.D and .E are also important for this section but were not repeated here.

(b) Section 2E, Item IV.E.6 - Before performing this step, the operator had to remember to invert the connection arrangement called out in step E.1, because it was not presented explicitly.

(c) Section 2A.IV - A discussion of EOI-14, Anticipated Transients Without Scram (ATWS), had not been included; yet, symptoms and action statements lead to the possible conclusion that an ATWS event was occurring.

d. AOI-3, Revision 3, Malfunction of Reactor Makeup Control

(1) Category I

(a) Sections 3A.IV.H, 3C.IV.D, and 3D.III.G - These sections appeared technically incorrect and might better read, "Equalize boron concentration in reactor coolant and pressurizer by manually increasing pressurizer spray flow. Backup heaters will come on automatically to maintain reactor cooling system (RCS) pressure."

e. AOI-4, Revision 4, Nuclear Instrumentation Malfunction

No comments

f. AOI-5, Revision 3, Operation with one Reactor Coolant Loop out of Service

(1) Category I

(a) Item 9.2 of the Technical Specifications forbids 3-loop operation until NRC gives permission.

(b) Discussion Section - The Statement that ". . . limited to a period no longer than the time required to plan the outage . . ." is not valid. The only valid guidance is the Technical Specification.

(c) Discussion Section - The correct Technical Specification section was not referenced.

(d) Discussion Section - Stated that "The unit load will be reduced to 15 percent power before pump shutdown." This will not be possible in all plant situations.

g. AOI-6, Revision 6, Excess Primary Plant Leakage

(1) Category I

(a) Section III.A - Pressurizer level may not decrease in certain accidents, such as pilot-operated relief valve (PORV) leaks.

(b) Figure 1 relied on pressurizer pressure or level as initial symptoms where many other symptoms were possible (see list on pages 1 and 2 of instruction).

(c) Section IV.A - This section stated the action to be taken if a PORV is leaking; however, it did not give instructions to be followed if a safety valve is leaking.

(d) Section IV.F - More than a "caution" statement seems warranted to address leaks in the auxiliary building.

(e) Figure 1 used decreasing pressurizer pressure and level for the beginning of the event tree, but the AOI was to address a leak small enough that one charging pump can maintain pressurizer level.

(f) Section IV.F - The caution following this step referred to appropriate radiation protective measures rather than giving definite instructions to refer to the Radiological Emergency Plan (REP).

(2) Category II

(a) Leakage and operator action when in residual heat removal (RHR) mode and leakage is into the auxiliary building were not addressed.

(b) Section V - The reference to FSAR section 16.3.4.6.1 is no longer valid since it has been replaced by the Technical Specifications.

(c) Section IV - The caution referred to Technical Specification limits instead of giving specific directions to the proper Technical Specification section or referring to the subsequent discussion part of the instruction where limits were provided.

h. AOI-7, Revision 2, Maximum Probable Flood

(1) Category II

(a) Revision 2 of Flood Design Criteria (SQN-DC-V-12.1) had not been incorporated into this instruction with the following results.

- ° No instructions had been provided for preparing sump pumps for the diesel generator building, essential raw cooling water (ERCW) intake station, and the spent fuel pit cooling system (SFPCS) pump enclosure.

- ° No instructions had been provided for closing drain valves or for installing blind flanges to prevent water entry to the diesel building, ERCW station, and SFPCS pump enclosure.

(b) The warning scheme for seismic floods was not consistent with that described in FSAR 2.4.A.9.

(c) The means for confirming that an official flood warning had been received were not presented or referenced.

(d) Item V.B.31 (and other corresponding sections) - FCV 30-40, -50, -52, -54, -56, and -58 were apparently overlooked in compiling the lists.

If the purge penetrations of the shield building had flexible connections on the exterior which may be submerged, then ducting in the annulus may have leakoffs which would allow drainage into the annulus. This section of the instruction should provide for closure of those leakoffs.

(e) Items V.F.26 and V.H.26 - The instruction did not require closing the valves which segregate the high pressure fire protection (HPFP) piping in the auxiliary building from the turbine building.

(f) Item V.B.21 - The words "control circuitry flow" need clarification.

i. AOI-8, Revision 3, Tornado Watch

(1) Category I

(a) Items IV.A.2 and IV.A.3 - These items would have to be rearranged to shut down the HVAC equipment prior to closing tornado dampers.

(b) Item IV - Securing turbine building cranes was addressed; however, auxiliary building cranes could also be a threat if not secured away from the spent fuel pool during a tornado warning.

(2) Category II

(a) It is important to verify that tornado doors and hatches are closed during a tornado warning.

(b) The control building pressurizing air intake dampers are subject to freezing in freezing weather and may need to be thawed (FSAR Q9.4-1).

(c) Section IV.E - A reference to, or repetition of, the recovery procedure EOI-5.V.Y to purge the diesel generator engines of accumulated combustible material following light-load operation had not been included.

j. AOI-9, Revision 2, Earthquake

(1) Category II

(a) Section II - This section contained no caution that spurious operation of systems, equipment, and instrumentation not seismically qualified can occur.

(b) Section IV - The sixth paragraph could give more specific information, such as, "The accelerometers are located (1) on top of panel 0-M-25 in the main control room, unit 1 auxiliary control building, (2) on the safety injection system piping at el. 702.0 in the unit 1 reactor building, and (3) on the upper head injection system piping at el. 706.75 in the unit 1 reactor building.

(c) Section IV - Similarly, the following discussion would fit between paragraphs 7 and 8:

An active triaxial response spectrum recorder is located in the unit 1 reactor building on the base slab at el. 679.78 in the annulus between the shield building and the containment vessel. The unit will record a number of maximum amplitude of acceleration with frequencies between 2 and 26 Hz and will trigger an alarm when any of the accelerations with frequencies 5.0, 6.4, 10.1, and 16.0 Hz exceed a preset amplitude.

A passive triaxial response spectrum recorder, identical to the one mentioned above with the exception that it is not equipped with alarm contacts, are in the following locations: (1) unit 1 reactor building on the floor slab at el. 733.63, (2) auxiliary control room on the floor slab at el. 734.0, and (3) diesel generator building on the base slab at el. 722.0.

k. <u>AOI-10, Revision 3, Loss of Control Air</u>

(1) <u>Category I</u>

(a) Section 10A, Item 3.0 - Another instruction was referenced in the immediate operator action section. While this makes AOI-10 more compact, it creates uncertainty as to the timeliness of the operators actions. NSRS recommends the immediate operator actions be self-contained if possible.

(b) Section 10A, Item 3.0 - Reference to AOI-1 was not correct since it was written assuming control air is available.

(c) Section 10.A, Item 4.D. - The steam dump valve will not have control air available.

(d) Section 6.0 - The list did not provide a complete status of all emergency core cooling systems (ECCS). For example, no heating, ventilating, and air-conditioning (HVAC) status was provided (transformer room cooling).

(2) Category II

(a) Section 10.A, Item 4.B.4 - The [illegible]rs and filters need not be bypassed. The installed sp[illegible] can be used instead, and flow indicators near the dryers can be used to determine whether the problem is upstream or downstream of the flow indicators.

(b) Section 10.C, Items 4.A.6 and 4.B.6 - The valves from control to auxiliary compressed air systems need not be opened since the auxiliary air compressors will be running to supply either auxiliary train which will hold pressure. Opening the valves after the damage is isolated or repaired is acceptable.

(c) Section 10.C, Item 5 - The ice condenser system is improperly listed as equipment supplied with auxiliary control air.

1. AOI-11, Revision 1, Loss of Condenser Vacuum

(1) Category I

(a) Complete loss of condenser vacuum will prevent use of turbine bypass. The instruction did not provide guidance for this condition.

(2) Category II

(a) Item I - Another symptom is condenser vacuum low alarm at 5.5 inches Hg ABS (2.7 psia) panel XA-35-2C-20.

(b) Item II.A - The automatic action was incorrectly stated. The turbine will trip (and the reactor will trip if above P-7) due to low condenser vacuum at 8 to 12 inches Hg abs (3.98 psia to 5.9 psia).

(c) Item IV - In addition to the actions listed, the operator may obtain main condenser air inleakage rates from rotameters on vacuum pumps and/or flow elements in condenser vent piping. If flows above are normal, check the following:

- Vacuum breaker closed.
- Gland seal water on hotwell pumps, vacuum breaker, etc.
- Seal water on vacuum pump.
- Large air inleakage into main condenser from failed joints or other components under vacuum such as MFPT condensers and feedwater heaters.
- Proper steam seal pressure at seals.
- Steam supply available to main steam relief (MSR) safety valves if starting unit.

m. AOI-12, Revision 1, Loss of Containment Integrity

(1) Category I

(a) Section 12A, Item III - Actions to be taken were not specified.

(b) Section 12A, IV.A - Technical Specification section number had not been specified.

(c) Section 12A, Item IV.B - Operator actions if radioactivity release had occurred were not given.

(d) Section 12B, Item IV.A - Corrective actions to be taken were not specified.

(2) Category II

(a) Section 12A, Item V, Discussion Section - This section could be rewritten for clarity and completeness to read as follows:

> The purpose of this instruction is to provide indications of and operator actions for loss of containment integrity.
>
> Containment integrity is defined to exist when all of the following are satisfied:
>
> 1. All penetrations required to be closed during accident conditions are either:
>
>   a. Capable of being closed by an operable containment automatic isolation valve system or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed position, except as provided in table 3.6.2 of Technical Specification 3.6.3.1.

2. All equipment hatches are closed and sealed

3. Each air lock is operable pursuant to Technical Specification 3.6.1.3.

4. The containment leakage rates are within the limits of Technical Specification 3.6.1.2.

5. The sealing mechanisms for all penetrations not discussed above are intact.

6. The emergency gas treatment system (EGTS) is operable and capable of maintaining the annulus pressure at -1/2 inch of water with an exhaust flow rate of 125 CFM.

7. The auxiliary building gas treatment system is operable and capable of maintaining the auxiliary building secondary containment enclosure at a pressure of -1/4 inch of water with an exhaust flow rate of 9000 cfm.

8. The annulus pressure is maintained at -5 inches of water when there is irradiated fuel in the reactor or primary containment and the reactor is in any mode other than cold shutdown.

9. Intermediate deck doors and top deck doors are closed.

10. Ice condenser inlet operable and air return fans operable.

11. The personnel access doors and equipment hatches between the containment upper and lower compartments shall be operable and closed.

12. The ice condenser floor drains shall be operable.

To better delineate containment requirements during the different anticipated plant conditions, this instruction is divided into the following sections:

A. Power Operation and Shutdown

B. Refueling Operation

(b) Section 12A, Item I.D - NSRS recommends that the following note be added to this section: "CAUTION - The equipment hatch and at least one door in each personnel air lock must be verified to be closed prior to placing the reactor in any condition other than cold shutdown."

(c) Section 12B, VI - The reference to FSAR section 16.3.9.4. is no longer valid.

n. AOI-13, Revision 2, Loss of Essential Raw Cooling Water

(1) Category I

(a) Continued unit operation after isolation of a main ERCW header in the auxiliary building is risky. Continued header operation is much preferred to isolation. Hence, "damage control" techniques, if investigated, might show where ERCW leaks might be patched without isolation until a convenient repair situation is available. This approach is allowed under Technical Specifications. The AOI could address temporary repair as a valid alternative to header isolation and possible immediate unit shutdown. The AOI should also address the condition where leakage continues until "damage control" techniques are implemented or the plant is in a configuration where ERCW header isolation is acceptable.

(b) Additional AOI directions are needed for the operator to suitably evaluate and control the effects of isolating cooled equipment from an ERCW header. The lists of equipment isolated on AOI-13, pages 11-14, include equipment for which:

(1) Damage is imminent unless turned off (e.g., centrifugal charging pump).

(2) Redundant counterpart must be operating to prevent equipment damage (e.g., control rod drive mechanism (CRDM) coolers).

(3) Multiple systems are affected through their loss (e.g., component cooling system (CCS) heat exchangers and control air system air coolers).

(4) Nothing is particularly significant (normally) about their loss during isolation of an ERCW header.

Suitable checks, cautions, and equipment trips could be prioritized and included with the equipment in categories (1) and (2). Additional instructions or reference to other suitable AOI's should be sufficient to handle the loss of equipment in category (3).

(c) The instruction listed and discussed unit 2 equipment, but it was applicable to unit 1 only.

this is of no consequence to the present SQN SRO's and RO's because they had been trained in the "cold" program which required more extensive simulator training than the "hot" program.

b. NRC Requirement - Eligibility requirements shall be developed for instructors, in addition to that listed in A.2 above.

NSRS Observation - TVA's training center instructors presently meet the requirements of a shift engineer (see item D.1 above). However, in the past this had not been a requirement to become an instructor. In addition, TVA was upgrading an existing program used to train the instructors in methods of teaching and communication techniques. Part-time instructors from SQN had not participated in the program.

3. <u>NRC Examinations</u>

a. NRC Requirement - NRC shall administer the certification examinations that are presently administered at the conclusion of the off-site portion of the cold training programs.

NSRS Observation - Examinations had been administered in the past by both T[illegible] training instructors and by the NRC at the conclusion of the cold license training program; therefore, the NRC's intent of this requirement is not clear.

b. NRC Requirement - All applicants shall be required to be administered a simulator examination in addition to the written examinations and plant oral tests.

NSRS Observation - Previously TVA RO and SRO applicants had taken a simulator examination for their license. During phases of their training, the applicants were tested and evaluated on the simulator by a training center instructor.

c. NRC Requirement - NRC shall administer the requalification program annual examination.

NSRS Observation - In the past TVA training instructors had administered the annual requalification examination.

(b) AOI-15B, Figures 1-7 - These figures were too large to reproduce entirely on the sheets and bear no evidence of EN DES review. Approved drawings should be used.

q. AOI-16, Revision 3, Loss of Normal Feedwater

(1) Category II

(a) Item II.B - "Steam generator blowdown will isolate," is an automatic action which is applicable here.

(b) Item III.A.3 - A more precise statement would be, "Verify steam dump is controlling average reactor coolant temperature (T_{avg}) within the range of 577° F to 547° F. Steam dump will eventually decrease T_{avg} to within a few degrees above 547° F as system conditions stabilize."

(c) Item IV.A.1 - The acoustic monitoring indicators are a valid means to verify reclosure of the power-operated relief and/or safety valves. Close block valve, if necessary.

(d) Item IV.E - "Verify that all turbine drains are open," is an appropriate phrase to add here.

(e) Item IV.F - The bearing lift pump starts at 600 RPM, not 900 RPM.

(f) Item V - Additional sources of water for the steam generator are:

6.	Condensate Booster Pumps (CBP's), Hot Well Pumps (HWP's), and Condensate Demineralizer Pumps (CDP's)	0-600 psig
7.	HWP's and CDP's	0-300 psig

r. AOI-17 Revision 4, Turbine and Generator Trips

(1) Category I

(a) Section III.A - It is not clear that the operator is obliged to return to this procedure after performing AOI-1 Immediate Operator Actions.

(2) Category II

(a) Item I.A.5 - The "65 psig" was incorrect. The correct pressure is "60 psig."

(b) Item I.A.7 - The " 5.5" Hg abs 2.7 psia decreasing" was incorrect. The correct pressure is "(5.0" Hg abs)."

(c) Item III.B.2 - The first sentence would be more precise as follows:

> T_{avg} is being controlled within the range of 557° F to 547° F by steamp dumps or if steam dumps are not available due to low condenser vacuum, indirectly by the atmospheric reliefs controlling the steam generator pressures to 1025 psig..

(d) Item III.B.3 - The following had been omitted from this section:

> "III.B.3 Generator breakers open, 6.9 kV unit station service transferred."

(e) Item IV.B - If a main feedwater pump (MFP) is available and the main condenser backpressure is below the steam dump block point, the reactor could continue to operate at the pretrip level (below P-7). At a higher reactor power level (5 to 8 percent), the main feedwater system would function better than at 2 percent power and also there would be less chance of overcooling a steam generator or the reactor.

(f) Item IV.B.6 - The bearing lift pumps start at 600 RPM instead of 900.

(g) Item IV.E.4 - The turbine drains need to be open as well as the main steam line drains.

s. <u>AOI-18, Revision 2, Malfunction of Pressurizer Pressure Control System</u>

(1) <u>Category II</u>

(a) Section 18A, Item I.A.1 - The alarm setpoint is 20° F.

(b) Section 18A, Item I.A.2 - The alarm setpoint is 20° F.

(c) Section 18A, Item I.A - Safety or relief valve open (from acoustic monitoring system) is also a symptom.

(d) Section 18A, Item I.C - The acoustic monitoring system will also indicate PORV or safety valves open.

(e) Section 18A, Item I.E - The pressurizer level may not drop.

(f) Section 18A, Item II - Another automatic action is safety injection on low pressurizer pressure.

(g) Section 18A, Item III.A - Inserting "(2315 psig)" after normal automatic closure is recommended.

(h) Section 18B, Item I.B.5 - A single channel failure will not cause PORV opening due to the interlock with other channel (setpoint of interlock is now 2335 psig). This comment also applies to items II.D, III.A, and III.C.

(i) Section 18B, Item III.B - Channel III (68-323) was listed incorrectly; channel II (68-334) is the correct channel.

(j) Section 18B, Item VI, Second Paragraph - The low pressure interlock setpoint is 2335 psig.

(k) Section 18C, Item II.B - The reactor trip occurs at 1970 psig.

(l) Section 18C, Item III.D - Instructions to terminate safety injection were not consistent with procedures developed by Westinghouse Owners Group. It is also necessary to monitor the margin to saturation.

(m) Section 18E, Item IV.D, Note - NSRS recommends adding the following: "Monitor margin to saturation on computer readout."

t. <u>AOI-19, Revisiion 4, Inadvertent Safety Injection</u>

(1) <u>Category I</u>

(a) Section III.A.4 - The second sentence covers "auxiliary feedwater pumps running" but did not require the operator to check that auxiliary feedwater system valves were in proper position.

(b) Westinghouse has issued Appendix A of Volume III of Westinghouse Topical Report WCAP-9600. Guidelines provided by this instruction were not utilized in development of AOI-19.

(c) Item IV.C - The sequence followed here did not assure necessary protection of the reactor coolant pump (RCP) seals by seal injection.

u. AOI-20, Revision 2, Malfunction of Pressurizer Level Control System

(1) Category II

(a) Item I.C.1 - An additional symptom is: "c. Backup heaters on if channel selector in position 1."

(b) Item I.C.2 - An additional symptom is: "b. Letdown isolation and heaters off if channel selector in position 3 or 1."

(c) Item II.C - Both steps under this item assumed that this LT had not been selected by XT-68-339E. If it had been selected, then actions added in comments on items I.C.1 and I.C.2 above would automatically occur.

v. AOI-21.1, Revision 3, Loss of 125-V DC Vital Battery Board 1

(1) Category II

(a) Item IV.F - If the EGTS train were put into operation, it would defeat the purpose of the annulus vacuum control system which is to maintain a negative pressure in the annulus.

(b) Item IV.F.4 - The EGTS and auxiliary building gas treatment system (ABGTS) will automatically start when called upon, so they do not have to be manually started unless needed.

w. AOI-21.2, Revision 3, Loss of 125-V DC Vital Battery Board II (Unit 1)

(1) Category II

See comments on AOI 21.1.

x. AOI's 21.3, Revision 3, and 21.4, Revision 3, Loss of 125-V DC Vital Battery Board III and IV; and AOI's 21.5, Revision 1, 21.6, Revision 1, 21.7, Revision 1, and 21.8, Revision 1, Unit 2 Vital Battery Boards

No comments.

y. AOI-22, Revision 1, Break of Downstream Dam

No comments.

z. AOI-23, Revision 1, Reactor Coolant Pump Seal Abnormalities

No comments.

a.1 AOI-24, Revision 0, Steam Generator Tube Leak

(1) Category I

(a) Westinghouse has issued Appendix A of Volume III of Westinghouse Topical Report WCAP-9600. Guidelines provided by this instruction had not been utilized for development of AOI-24.

APPENDIX B

CRITERIA FOR REACTOR OPERATOR

TRAINING AND LICENSING

A. <u>Eligibility Requirements to be Administered an Examination</u>

1. <u>Experience</u>

a. NRC Requirement - Applicants for senior operator license shall have four years of responsible power plant experience. Responsible power plant experience should be that obtained as a control room operator (fossil or nuclear) or as a power plant staff engineer involved in the day-to-day activities of the facility, commencing with the final year of construction. A maximum of two years power plant experience may be fulfilled by academic or related technical training, on a one-for-one time basis. Two years shall be nuclear power plant experience. At least six months of the nuclear power plant experience shall be at the plant for which he seeks a license. This requirement is effective to applications received on or after May 1, 1980.

NSRS Observation - TVA's training program stated that an SRO should have five years of responsible power plant experience, of which a minimum of one year should be nuclear plant experience. Consequently, TVA's written program does not meet the new NRC requirements of two years nuclear plant experience with at least six months at the plant for which he seeks a license; however, the actual experience level of the SQN SRO's does meet the requirement.

NRC states that precritical applicants will be required to meet unique qualifications designed to accommodate the fact that their facility had not yet been in operation; however, these "unique qualifications" are not specified.

b. NRC Requirement - Applicants for senior operator licenses shall have held an operator's license for one year. This requirement is effective for applications received after December 1, 1980.

NSRS Observation - TVA's training program does not require SQN RO's to hold their license for a year before applying for an SRO license.

2. <u>Training</u>

a. NRC Requirement - Senior Operator: Applicants shall have three months of shift training as an extra man on shift. This requirement is effective for applications received after August 1, 1980.

NSRS Observation - TVA had not required SRO license applicants to have three months of shift training as an extra man on shift.

b. NRC Requirement - Control Room Operator: Applicants shall have three months training on shift as an extra person in the control room. This requirement is effective for applications after August 1, 1980.

NSRS Observation - TVA had not required RO license applicants to have three months of training on shift as an extra person in the control room. TVA has required personnel in training to have four weeks of on-shift training. The TVA program is being upgraded to meet the new requirements.

c. NRC Requirement - Training programs shall be modified, as necessary, to provide:

(1) Training in heat transfer, fluid flow and thermodynamics.

(2) Training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged.

(3) Increased emphasis on reactor and plant transients.

The revised programs should be submitted to NRC review by August 1, 1980.

NSRS Observation - (1) The NSGPO initial training program had included training in heat transfer, fluid flow, and thermodynamics.

(2) Degraded core conditions had not been a part of TVA's training program, but the training program was being updated to address this requirement.

(3) All phases of operator training had emphasized reactor and plant transients during the study of emergency and abnormal operating instructions.

d. NRC Requirement - Training center and facility instructors who teach systems, integrated responses, transient and simulator courses shall demonstrate their competence to NRC by successful completion of a senior operator examination. Applications should be submitted to the NRC no later than August 1, 1980, for individuals who do not already hold a senior operator license.

NSRS Observation - All TVA training center and facility instructors who teach systems, integrated responses, transient, and simulator courses hold an SRO license at SQN.

e. NRC Requirement - Instructors shall be enrolled in appropriate requalification programs to assure they are cognizant of current operating history, problems, and changes to procedures and administrative limitations. Programs should be initiated by May 1, 1980, and submitted to NRC for review by August 1, 1980.

NSRS Observation - To maintain an SRO license and to assure and maintain an up-to-date operating plant knowledge, each training center instructor had been required to participate in the appropriate requalification program.

3. Facility Certifications

NRC Requirement - Certifications completed pursuant to sections 55.10(a)(6) and 55.33a(4) and (5) of 10CFR Part 55 shall be signed by the highest level of corporate management for plant operation (for example, Vice President for Operations). This requirement applies to applications received on or after May 1, 1980.

NSRS Observation - The plant superintendent and the director of the Division of Nuclear Power have certified all operator licenses.

B. NRC Examinations

1. Increased Scope of Examinations

a. NRC Requirement - A new category shall be added to the operator written examination entitled, "Principles of Heat Transfer and Fluid Mechanics." This requirement will be effective for NRC examinations administered after May 1, 1980.

NSRS Observation - TVA was adding the study of heat transfer and fluid mechanics to the RO/SRO training program to prepare applicants for the new categories on the NRC examinations.

b. NRC Requirement - A new category shall be added to the senior operator written examination entitled "Theory of Fluids and Thermodynamics." This requirement will be effective for NRC examinations administered after May 1, 1980.

NSRS Observation - TVA's training program was also being revised to cover the theory of fluids and thermodynamics to prepare applicants for the new categories on the NRC examinations.

c. NRC Requirement - Time limits shall be imposed for completion of the written examinations:

(1) Operator: 9 hours

(2) Senior Operator: 7 hours

This requirement will be effective for NRC examinations administered after May 1, 1980.

NSRS Observation - TVA did not have a requirement in the training program to limit the time for completion of the NRC written examination. No time limit had been set by the NRC regulations in the past.

d. NRC Requirement - The passing grade for the written examination shall be 80 percent overall and 70 percent in each category. This requirement will be effective for NRC examinations administered after May 1, 1980.

NSRS Observation - In the SQN RO/SRO training program, TVA had applied the NRC criteria of 80 percent overall and at least 70 percent in each category of the examination; however, the written program had not yet been revised to reflect the new criteria.

e. NRC Requirement - All applicants for senior operator licenses shall be required to be administered an operating test as well as the written examination. This requirement will be effective for NRC examinations administered after May 1, 1980.

NSRS Observation - TVA RO and SRO candidates had taken an NRC simulator licensing examination. During phases of their training, the applicants were tested and evaluated on the simulator by training center instructors.

f. NRC Requirement - Applicants will grant permission to NRC to inform their facility management regarding the results of the examinations for purposes of enrollment in requalification programs. This requirement is effective for applications received after May 1, 1980.

NSRS Observation - In the past an applicant had to request his exact score since the NRC only informed the applicant and his company if he had passed or failed the licensing examination. Because this information had not been available to TVA, all RO's and SRO's were entered in a general requalification program. If a weak area was discovered as a result of the requalification examination administered by TVA, then the operator was placed in an accelerated program to cover the deficient subject.

C. Requalification Programs

1. NRC Requirement - Content of the licensed operator requalification programs shall be modified to include instruction in heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core. This requirement is effective May 1, 1980.

   NSRS Observation - TVA had added some discussion of heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core to the requalification training program; however, the program may require additional information on these subjects in light of the more detailed course discussion included in the NRC's March 29, 1980, letter.

2. NRC Requirement - The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license; 80 percent overall and 70 percent each category. This requirement was effective as of March 29, 1980.

   NSRS Observation - TVA's program required operators to score at least 70 percent overall on the requalification examination to allow him to continue in his licensed activities. However, if an operator scored less than 80 percent in any category of the written examination, he was required to attend additional lectures on that particular subject. Consequently, TVA's written program did not meet the new criteria; however, TVA is committed to apply the new grading criteria to the current participants of the requalification program at SQN.

3. NRC Requirements - Programs should be modified to require the control manipulations listed in Enclosure 4 of the modified NRC criteria. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operations must be walked through with, and evaluated by, a member of the training staff at a minimum. An appropriate simulator may be used to satisfy the requirements for control manipulations. This requirement specifies programs to be modified by August 1, 1980, and will apply to reuewal applications received after November 1, 1980.

NSRS Observation - TVA's program had not required documentation that the operator had performed all of the control manipulations listed in Enclosure 4 of the NRC's March 29, 1980, letter.

D. Long-Range Criteria and/or Requirements

The following require additional NRC staff work and/or rulemaking prior to their implementation.

1. Qualifications

a. NRC Requirement - Shift supervisors shall have an engineering degree or equivalent qualifications.

NSRS Observation - The shift engineer had not been required to have an engineering degree. However, they had been required to have an SRO license or six years of responsible power plant experience with at least one year in a large nuclear plant.

b. NRC Requirement - Senior operators shall have successfully completed a course in appropriate engineering and scientific subject equal to 60 credit hours of college level subjects.

NSRS Observation - SRO's had not been required to complete 60 hours of college level engineering or scientific subjects. However, each SRO had been required to have completed certain operator training requirements and to have five years of responsible power plant experience with at least one year in a large nuclear plant. In addition, the Division of Nuclear Power is working on accreditation of the NSGPO program such that this requirement will be met.

2. Training

a. NRC Requirement - All applicants shall attend simulator training programs. Required control manipulations and exercises to be performed shall be the same for "cold" and "hot" applicants.

NSRS Observation - TVA's written program had not required the control manipulations and exercises to be the same for "cold" and "hot" candidates; however, this is of no consequence to the present SQN SRO's and RO's because they had been trained in the "cold" program which required more extensive simulator training than the "hot" program.

b. NRC Requirement - Eligibility requirements shall be developed for instructors, in addition to that listed in A.2 above.

NSRS Observation - TVA's training center instructors presently meet the requirements of a shift engineer (see item D.1 above). However, in the past this had not been a requirement to become an instructor. In addition, TVA was upgrading an existing program used to train the instructors in methods of teaching and communication techniques. Part-time instructors from SQN had not participated in the program.

3. <u>NRC Examinations</u>

a. NRC Requirement - NRC shall administer the certification examinations that are presently administered at the conclusion of the off-site portion of the cold training programs.

NSRS Observation - Examinations had been administered in the past by both T. training instructors and by the NRC at the conclusion of the cold license training program; therefore, the NRC's intent of this requirement is not clear.

b. NRC Requirement - All applicants shall be required to be administered a simulator examination in addition to the written examinations and plant oral tests.

NSRS Observation - Previously TVA RO and SRO applicants had taken a simulator examination for their license. During phases of their training, the applicants were tested and evaluated on the simulator by a training center instructor.

c. NRC Requirement - NRC shall administer the requalification program annual examination.

NSRS Observation - In the past TVA training instructors had administered the annual requalification examination.

4. Requalification Programs

NRC Requirement - All licensees shall participate in simulator programs as part of the requalification programs. Control manipulations shall be performed pursuant to Enclosure 4.

NSRS Observation - TVA's requalification program included 40 hours of simulator training. However, all of the control manipulations stated in Enclosure 4 of the NRC's March 29, 1980, letter are not performed during the requalification program.

E. Conclusions

Subsequent to the TMI-2 incident, the NRC had undertaken extensive study in the area of operator training and qualifications. As a result of these studies, new requirements have been developed, and new interpretations were applied to old requirements. The NRC had pursued implementation of some of the requirements through issuance of orders, bulletins, and licensee commitments as conditions for the issuance of an operating license. This criteria for reactor operator training and licensing is an accumulation of these requirements and summarizes the NRC intent of implementation. NSRS finds that TVA's existing documents training and requalifications were written to meet existing NRC requirements and generally meet these requirements with the few exceptions noted in this report. TVA's existing program does not meet the new requirements in many areas as identified in this appendix; however, the TVA program is being upgraded in good faith to meet many of these rapidly changing NRC requirements. NSRS also finds that, although SQN RO's/SRO's could meet some of the new criteria, the majority of the criteria was not applied to the SQN RO's and SRO's. Since SQN has at present a full complement of RO's and SRO's who had just completed the requalification training, the new criteria regarding new RO's and SRO's and the new criteria relating to the requalification training are not of immediate concern.

DES 791011 031

G. H. Kimmons, Manager of Engineering Design and Construction, W12A9 C-K
H. G. Parris, Manager of Power, 500C CST2-C

Charles Bonine, Jr., Manager of Management Services, W12A1 C-K

OCT 5 1979

WATTS BAR NUCLEAR PLANT - EMPLOYEE CONCERN OVER EXCESSIVE WIRE LIFTING TO FACILITATE TESTING 79-09-01

Attached is the Nuclear Safety Review Staff's report on employee concern 79-09-01. The report makes recommendations which close the employee concern if they are implemented. If you do not concur with the recommendations in the report, please let me know by November 15, 1979. Otherwise, we will consider the employee concern closed. The Nuclear Safety Review Staff will informally monitor the implementation of the recommendations.

Charles Bonine, Jr.

EGB 10/1/79
EGB:KRW
Attachment
cc: E. A. Belvin, ROB-M (Attachment)

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFTEY REVIEW STAFF

Employee Concern - Case No. 79-09-01

Subject: TVA
Watts Bar Nuclear Plant

Employee Concern over Excessive Wire Lifting Practices to Facilitate Testing

Period of Investigation: September 17 - September 24, 1979

Investigator: T. G. Tyler
T. G. Tyler — 9-28-79 Date

Reviewed by: H. L. Jones
H. L. Jones — 9/28/79 Date

H. E. McConnell
H. E. McConnell — 9/28/79 Date

Approved by: E. G. Beasley
E. G. Beasley, Chief, Nuclear Safety Review Staff — 10/1/79 Date

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I. Case 79-09-01 Concern

On Monday, September 17, 1979, the Nuclear Safety Review Staff received the following employee concern from Mr. Elmer Todd of the Division of Power Systems Operations (DPSO).

> During preparation of maintenance and surveillance instructions that cover DPSO testing of protective relays, meters, indicating instruments, and transducers on the 480-volt shutdown boards, 6900-volt shutdown boards, and diesel generator boards, the employee realized that these electrical boards were not equipped to facilitate the required testing without having to lift leads, install test leads, remove test leads, and reterminate the board wiring. In some instances as many as 50 wires all the same color (gray) will have to be lifted and reterminated. He considered such a practice to be unsafe in that the potential exists for making retermination errors that could result in protective relaying misoperations, open-circuiting current transformer (CT) circuits which could cause the CT to fail and possibly damage primary phase conductors, and/or short-circuiting or backfeeding potential transformer (PT) secondaries which could pose hazards to personnel and operating integrity.
>
> The employee stated that over a year ago EN DES was requested via a DCR to install test blocks in order to eliminate the wire lifting practice, and that EN DES had refused to install the test blocks on the basis that the frequency of testing these components did not justify installation of the test blocks. The employee was not satisfied with this situation and would have taken his concern to the NRC had the appeals process in the Nuclear Safety Review Staff not been established.

II. Conclusions

1. Mr. Todd's concerns are valid and express a commendable attitude on behalf of DPSO and Division of Nuclear Power personnel whereby detailed written maintenance instructions have been prepared to cover the testing of all components located on safety-related boards.

2. The principal factor relating to the employee's concern was a failure of the organizations involved to communicate effectively. The Division of Nuclear Power neither explicitly and completely described what testing capabilities they required nor completely explained why such capabilities were needed in the DCR's that were prepared. As a result the Division of Engineering Design (EN DES) never fully realizing what was needed or why it was needed disapproved the DCR's.

3. There were no indications that there was any unsafe attitude on the part of any individual interviewed. All parties recognized the need for these testing capabilities when the problem was fully explained.

III. Recommendations

1. The Division of Nuclear Power should resubmit DCR's with either sufficient verbal descriptions or sufficient marked-up prints to explicitly define where test blocks are needed to facilitate testing and maintenance of components on the 6900-volt and 480-volt shutdown boards and the diesel generator boards. Also the Division of Nuclear Power should designate in the DCR a point contact who is knowledgeable about what test capabilities are required in order to assist EN DES in the design of the testing provisions.

2. EN DES must make provisions to facilitate the testing, trouble shooting, and maintenance of the relays, meters, indicating instruments, transducers, and timers on the diesel generator logic and control boards, 6900-volt shutdown boards, and 480-volt shutdown boards.

3. From a generic standpoint, the Division of Nuclear Power should review maintenance and surveillance instructions at all nuclear plants to insure that the TVA policy of minimizing the number of jumpers and/or wire lifts required to facilitate testing or maintenance is being implemented. In cases where the instructions require wire lifts and/or jumpers, the Division of Nuclear Power should prepare and submit DCR's to EN DES that explicitly describe where and why test blocks are needed to be installed.

IV. Scope of Investigation

The scope of the investigation included the following:

A. Interviewed five (5) Division of Nuclear Power, four (4) Division of Power Systems Operations (DPSO), and four (4) Division of Engineering Design (EN DES) personnel.

B. Reviewed the maintenance instructions prepared by DPSO for the testing of protective relays, meters, indicating instruments, and transducers on the 6900-volt and 480-volt shutdown boards.

C. Reviewed marked-up drawings that are exemplary of the wire lifts required to facilitate testing in the 480-volt shutdown boards, 6900-volt shutdown boards, and diesel generator control boards.

D. Observed and photographed the as-installed hardware on the 480-volt shutdown boards, 6900-volt shutdown boards, diesel generator control boards, and diesel generator load logic panels.

V. Details of Investigation

A. Meeting with DPSO and WBN Plant Staff

The investigation into Case 79-09-01, employee concern over excessive wire lifting practices to facilitate testing, began with a meeting on September 20, 1979, at Watts Bar Nuclear Plant (WBNP) among the following:

R. L. Bruce - DNP[1]
R. D. Greer - DNP
J. T. Maddux - DNP
J. T. Groves - DNP
G. L. Williams - DNP
F. W. Bryan - DPSO[2]
D. C. Iles - DPSO
E. Todd - DPSO
W. D. King - DPSO
H. E. McConnell - NSRS[3]
T. G. Tyler - NSRS

The key points of the meeting are as follows:

1. During preparation of the maintenance instructions for the components on the 6900-volt and 480-volt shutdown boards and the diesel generator load logic and control boards, the WBN electrical maintenance staff along with DPSO personnel adopted a philosophy to write maintenance instructions that covered the testing of all components on a particular board rather than conform to the practice of writing a separate maintenance instruction for each type of component that was installed on a board. Subsequently, WBN wound up with three maintenance instructions for testing components on these boards whereas Sequoyah required the use of approximately one hundred instructions to perform the same testing.

2. While preparing these instructions in the first half of 1978, the DPSO personnel discovered that sufficient testing provisions were not present on these boards to facilitate testing without having to engage in an exceedingly large number of wire lifts, test lead installations, and wire reterminations. (Such a practice is contrary to TVA's position on the minimization of the use of jumpers and wire lifts as conveyed to the NRC in the answer to WBN FSAR question 040.10.). (Reference No's 7, 8 and 9)

3. Upon discovery of this condition the plant electrical maintenance personnel prepared WBNP Design Change Request WB-DCR-50 which was sent to EN DES on June 27, 1978, (DES 780628016). (Reference No. 5)

4. WB-DCR-50 was disapproved by EN DES on July 28, 1978, on the basis that the two year testing interval for these components was too infrequent to justify installation of test blocks. (Reference No. 5)

5. WBNP Design Change Request WB-DCR-74 was subsequently prepared and resubmitted to EN DES on May 4, 1979, (DES 790507022) again requesting installation of test blocks to facilitate testing on these boards. (Reference No. 6)

1. DNP - Division of Nuclear Power
2. DPSO - Division of Power Systems Operations
3. NSRS - Nuclear Safety Review Staff

6. WB-DCR-74 was disapproved by EN DES on September 5, 1979, on the basis of infrequent test intervals and existance of alternate isolation capabilities for testing voltmeters. (Reference No. 6)

7. The EN DES disapproval of WB-DCR-74 prompted Mr. Todd to bring this concern before the Nuclear Safety Review Staff as an open appeal.

8. A similar situation was discussed concerning the lack of sufficient testing capabilities to allow testing of the diesel generator load sequence timers without wire lifts. WB-DCR-44 was submitted to EN DES on March 30, 1978, (DES 7804004) requesting installation of testing capabilities for these timers. EN DES rejected WB-DCR-44 on April 27, 1978, (SWP 780427051) after telephone conversations with WBN personnel, on the basis that sufficient input was not provided by P PROD for EN DES to develop the best testing provisions to satisfy P PROD's testing needs. P PROD was requested in the EN DES rejection memo to review their requirements and resubmit the DCR. (Reference No. 4)

9. Marked-up prints depicting the worst cases for wire lifts to facilitate testing were reviewed.

10. The as-installed hardware on the 480-volt shutdown boards, 6900-volt shutdown boards, diesel generator load sequence panels, and the diesel generator local and main control room panels was observed and photographed. (Note: It was observed that the local diesel generator control boards supplied by the diesel manufacturer were equipped with test blocks to facilitate testing.)

11. The supervisor of the WBN instrumentation maintenance section was also interviewed with regard to the use of jumpers and wire lifts in maintenance and surveillance instructions for safety system logic and instrumentation. We (the investigators) were assured that the majority of the plant safety system logic and instrumentation was supplied with an adequate amount of testing capabilities or DCR's had been issued by P PROD and approved by EN DES to provide such testing capabilities.

B. Meeting with EN DES Personnel

As a continuation of the investigation into Case 79-09-01, a meeting was held on September 24, 1979, among the following:

H. B. Rankin - EN DES [1]
J. H. Boehms - EN DES
R. C. Denney - EN DES
W. M. Roop - EN DES
H. E. McConnell - NSRS [2]
T. G. Tyler - NSRS

1. EN DES - Divison of Engineering Design
2. NSRS - Nuclear Safety Review Staff

The key points of the meeting are as follows:

1. The Nuclear Safety Review Staff (NSRS) investigators reviewed the employee concern and the highlights of the meeting at WBNP on September 20, 1979, with those attending.

2. EN DES's reasons for disapproving WB-DCR's -44, -50, and -74 listed below were discussed:

   a. The DCR's did not contain sufficient information (i.e., all locations, sufficient description of how test provisions needed to be installed, how WBNP and DPSO personnel intended to test components, etc.) for EN DES to develop a design that they believed would satisfy the plant's testing needs.

   b. EN DES was unsuccessful in obtaining additional information on what P PROD wanted in the DCR's via several phone conversations with WBNP personnel.

   c. The full gambit of what the test blocks would be used for was never adequately expressed to EN DES. Subsequently, the frequency of testing argument was used by EN DES in their disapproval of the DCR's.

3. Criterion for inclusion of testing provision on switchgear, control boards, and logic panels were discussed. This discussion resulted in the following criterion:

   a. Routine testing or maintenance shall be capable of being performed within the technical specification outage time for the boards and/or panels without the installation of test capabilities.

   b. Hazard to personnel if test capabilities are not provided.

   c. Hazard to safety-related equipment if test capabilities are not provided.

   d. Frequency of testing and/or maintenance of components on these boards and panels.

4. EN DES explained that these same concerns are being partially addressed at plants later than WBNP as follows:

   a. At Bellefonte and Hartsville/Phipps Bend Nuclear Plants, test blocks are being provided on medium and low voltage switchgear in the current transformer circuits. No test blocks are provided in the potential transformer circuits; however, the voltmeters and voltage transducers can be calibrated after removing the fuses in the secondary of the potential transformers.

   b. Test blocks will be provided in both potential and current circuits on all medium and low voltage switchgear at Yellow Creek Nuclear Plant.

5. EN DES agreed to work with the Division of Nuclear Power in providing the needed test capabilities to facilitate testing and maintenance of components on these boards.

VI. References

1. Memo from Robert L. Wall to Henry A. Kyle, Jr., dated November 8, 1978, subject, "Installation of Test Facilities for Indicating Instruments and Transducers at the Nuclear Plants."

2. Memo from J. L. Thompson to F. W. Chandler dated November 24, 1978, subject, "Installation of Test Facilities for Indicating Instruments and Transducers at the Nuclear Plants," (EEB 781204012).

3. Memo from Roy H. Dunham to C. E. Winn dated December 22, 1978, subject, "Installation of Test Facilities for Indicating Instruments and Transducers at the Nuclear Plants," (EEB 781221934).

4. WBN DCR No. 44 and all supporting documentation[1] (DES 780404004).

5. WBN DCR No. 50 and all supporting documentation[1] (DES 780628016).

6. WBN DCR No. 74 and all supporting documentation[1] (DES 790507022).

7. WBN FSAR Question 040.10 and its response as added to the FSAR under Amendment WBNP-28.

8. WBNP Maintenance Instruction MI-57.50 Rev. 0, "6900-Volt Shutdown Boards Periodic Testing of Protective Relays, Meters, Indicating Instruments, and Transducers."

9. WBNP Maintenance Instruction MI-57.51 Rev. 0, "480-Volt Shutdown Boards Periodic Testing of Protective Relays, Indicating Instruments."

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1. Supporting documentation includes drawings, transmittal memos, and DCR disapproval memos.

Report 79-10-03

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : Richard L. Keck, Quality Assurance Engineer, W12B49 C-K

FROM : E. Gray Beasley, Acting Chief, Nuclear Safety Review Staff, 309 GB-K

DATE : October 11, 1979

SUBJECT: COMMENTS ON NUCLEAR POWER SAFETY FOR IEEE SPECTRUM MAGAZINE

We received a copy of your magazine article and interview comments to review from R. B. Connelly (ERI 791004 001). In reviewing the material and in a subsequent conversation between you and H. E. McConnell of our staff, we have concluded that your primary concern on nuclear power safety relates to a need for reliability engineering, especially in electrical power and control systems.

In its Nuclear Power Review Report, TVA discusses the need for improved data collection and failure reporting at all nuclear plants. The P PROD-EN DES task force on availability and reliability has been assigned responsibility for this item and has developed a schedule for implementation. The system will be implemented at Browns Ferry by the spring of 1980, and the task force will take the following actions to provide a thorough review of this material.

1. Identify by February 1, 1980, one specific organization to monitor and plot trends in equipment histories for all TVA nuclear plants and to compare these trends with data available outside.

2. Identify by February 1, 1980, the specific equipment whose histories should be reviewed in order to recognize potential safety and radiation problems.

3. Identify by March 1, 1980, the specific organizations which will review the equipment histories on the selected equipment.

While these actions alone do not assure reliability, they are an important part of TVA's overall availability and reliability program. The Nuclear Safety Review Staff is independently monitoring this program to verify that the objectives stated in the Nuclear Program Review have been accomplished.

We appreciate your interest in nuclear safety and thank you for directing our attention to this problem. If this response does not resolve your concern with respect to TVA's actions or if you have additional concerns, you should send us a written note describing them.

H. E. McConnell
for E. Gray Beasley

EGB:HEM:KRW
cc: E. A. Belvin, ROB-M

10/11/79 - EAB:JK
cc: C. Bonine, Jr., W12A1 C-K

10/11/79 - CB:CB
cc: G. H. Kimmons, W12A9 C-K

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : Richard F. Keck, Quality Assurance Engineer, W12D49 C-K

FROM : E. Gray Beasley, Acting Chief, Nuclear Safety Review Staff, 249A HBB-K

DATE : November 9, 1979

SUBJECT: EMPLOYEE CONCERN FOR RELIABILITY

Reference is made to your informal note of October 11, 1979, to H. E. McConnell of this staff and to your endorsement of October 15, 1979, on my memorandum of October 11, 1979, to you. In the informal note you referred to the question-and-answer section on page 9 of the June 1979 issue of Onsight. Mr. Mull's response is still the preferred method of handling such concerns, but the policy was changed to allow the employee to appeal directly to the director of the Office of Health and Safety in the event he has some concern about using the preferred method.

The new TVA policy on employee concerns is spelled out in the draft TVA Code on Differing Staff Opinions. The Code has been coordinated with the various divisions and offices and has been submitted to the General Manager for Board approval. The policy has been in effect a couple of months even though the formal approval was not completed.

In your endorsement of October 15, 1979, you expressed four concerns and recommended that an outside consultant be obtained to review TVA's reliability program. I am treating these as official concerns that come under the Code for Differing Staff Opinions, even though you may not have intended that they be handled that way.

NSRS is well aware of TVA's efforts in reliability through our contacts with OEDC and POWER and the NSRS monitoring of the action items in the TVA Nuclear Program Review. With that background I elected not to conduct our usual NSRS investigation of employee concerns and will respond to your concerns without further investigation.

For the past several years TVA has had an active reliability program; both EN DES and the former P PROD has had both a steering committee and a working level task force. Both of these groups have been actively working for the last couple of years. These two divisional efforts have worked closely with qualified reliability persons from organizations such as EPRI, DOE, EEI, NERC, NPRDS and with private organizations such as KAMAN and Mechanics Research. In addition, there have been detailed exchanges with other utility organizations. There have been joint programs with several of these organizations on collecting data and investigating various systems. Also, TVA has been an associate member of the UKAEA Systems Reliability Service for over two years and has access to the SRS data bank and counsel with SRS reliability engineers.

Richard F. Keck
November 9, 1979

EMPLOYEE CONCERN FOR RELIABILITY

The TVA Nuclear Program Review approved by the Board on June 1 of this year stresses reliability, establishes a data base, and utilizes our nuclear power experience in general. This has accelerated the TVA reliability effort. Certainly further acceleration is desirable, but it is somewhat constrained with the availability of resources and data.

It appears we should give the present program time to develop before undertaking further action.

E. Gray Beasley

E. Gray Beasley

EGB:KRW
cc: E. A. Belvin, ROB-M — Keck memo att.

11/9/79 - EAB:JK
cc: C. Bonine, Jr., W12A1 C-K — Keck memo att

TVA 64 (OS-9-68)

UNITED STATES GOVERNMENT

Memorandum

GNS 810410 050

TENNESSEE VALLEY AUTHORITY

TO : M. N. Sprouse, Manager of Engineering Design, W11A9 C-K
H. H. Mull, Manager of Construction, E7B24 C-K

FROM : H. N. Culver, Director of Nuclear Safety Review Staff, 249A HBB-K

DATE : April 9, 1981

SUBJECT: WATTS BAR NUCLEAR PLANT - NSRS REPORT R-80-21-WBN

The NSRS Report on the review of the WBN construction project program governing the installation and inspection of safety-related piping and supports conducted December 10-12, 1980, is attached for your information and actions on recommendations. Contained in the report are three recommendations that require action from your organizations to resolve. Other problem areas identified in the report are being addressed by OEDC in the Phase I and Phase II hanger and piping inspection programs and in response to NSRS as a result of reviews relating to IE Bulletin 79-14.

This report and its findings were reviewed with Messrs. Pierce, Cantrell, Killian, and Wilkins of your staffs in a meeting on March 11, 1981.

Please provide us a schedule for implementation of these recommendations by April 24, 1981. The NSRS contact for this report is T. G. Tyler, extension 6590.

H. N Culver

H. N. Culver

MVS

TGT:LML
Attachment
cc: MEDS, E4B37 C-K (Attachment)

READING FILE

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS Report No. R-80-21-WBN

Subject: Review of the WBN CONST Project Program Governing the Installation and Inspection of Safety-Related Piping and Supports

Date of Review: December 10-12, 1980

Reviewer: Terry G. Tyler 4/8/81
T. G. Tyler Date

Reviewer: Bruce F. Siefken 4/8/81
B. F. Siefken Date

Approved by: M V Sinkule 4/8/81
Date

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I. Introduction

During the period December 10-12, 1980, two members of the Nuclear Safety Review Staff (NSRS) conducted a review at Watts Bar Nuclear Plant (WBN). The objectives of the review were:

1. To develop an understanding of the way that the WBN Construction Project (CP) conceptulized, developed, implemented and revised the program that has controlled the installation of safety-related piping and supports. Inherent in this objective was the need to not only understand how the WBN CONST Project fulfilled their responsibilities in this program, but also to understand how the quality assurance (QA) organizations (WBN CONST Site, CONST QA Knoxville, and OEDC QA) and the EN DES branches and design project interacted with this program. The goal of the reviewers was to develop a detailed understanding of the program from identification of the design requirements by the CONST project to documentation of the final "as-constructed" system configuration including problems any person(s)/or groups of persons were having or had with understanding/implementing the program.

2. To determine whether or not the WBN program governing the installation of safety-related piping and hangers contained problems of a similar nature to those found to exist at Sequoyah during the NSRS review of the TVA Program to meet the requirements of IEB 79-14.

3. To make recommendations on ways to resolve significant nuclear safety-related problems that were found during the review and/or to make recommendations on ways to improve the methods and practices that currently govern the installation and inspection of safety-related structures, systems, and components at all of TVA's nuclear plants.

The review was accomplished by conducting individual discussions with managers from the WBN CP organization at the project manager, construction engineer, assistant construction engineer, section supervisor, quality assurance, craft superintendent, and assistant craft superintendent levels and with employees in welding, quality control and records, and crafts. This allowed NSRS to obtain a perspective of the safety-related piping and supports installation and inspection program at each level of the WBN CP organization. It also allowed the WBN CP Staff the opportunity to express their viewpoint of the program including its weaknesses, adequacy, problems encountered with implementation, and suggestions for improvement.

Prior to the actual review the NSRS representative had reviewed the NRC, EN DES, OEDC QA, CONST QA, and WBN CP generated documentation listed in Section V of this report. In addition, conversations had transpired between NSRS and Civil Engineering Branch (CEB) personnel and Nuclear Engineering Branch (NEB) personnel on various aspects of General Construction Specification G-43, "Support and Installation of Piping Systems in Category I Structures," This provided an understanding of the written requirements for such a program including commitments and a basis for determining the adequacy of the written program. From this review specific questions were generated which served as the basis for the onsite review interviews. The NSRS report on the "TVA Program to Meet IE Bulletin 79-14" (GNS 800814 001) and the two OEDC responses to that report--(1) EDC 800827 020 outlining a two-step program to ensure consistency between the "as-designed" and "as-constructed" safety-related piping and hanger configuration and (2) EDC 801212 013, in draft form at that time, outlining management control improvements for the quality aspects of the piping and supports program--were reviewed for problem similarities and committed programmatic revision applicability at WBN.

This report is for the most part founded on information that was revealed in the discussions that took place during the onsite review. As such, report readers should recognize that changes to current WBN CONST project practices and philosophies are in the process of implementation. NSRS views these changes as positive steps toward resolving some of the problems identified during this review. The overall effectiveness of these changes cannot be determined until after the passage of time.

II. SUMMARY

Assessment of Safety-Related Piping and Supports Program

The majority of the safety-related piping is installed in both units at WBN, while the majority of permanent hangers and supports remain to be installed. Considerable difficulty is being experienced with installing permanent hangers and supports per design requirements. The hanger and support installation problems appear to be due to piping being mislocated as a result of inadequate control by the WBN CP over the preparation of field fabrication sketches by the craft; to interferences with field routed components and structures; and to the use of different reference points by EN DES to locate piping, hangers, and supports.

The quality control program that governs the installation and inspection of safety-related piping, supports, and hangers appears to suffer from problems similar to those identified in NSRS's report on the "TVA Program to Meet IE Bulletin 79-14" for SQN. The Phase I and Phase II program to meet NRC IE Bulletin 79-14 (EDC 800827 020) and the management changes described in EDC 801212 013 should help to correct these problems. However, the passage of some amount of time is required before the effectiveness of these changes can be assessed.

The review revealed that the requirements contained in the quality control program constitute the best effort of the WBN CP to identify and interpret program requirements from the morass of documents containing design requirements, program requirements and licensing commitments that were provided to them by OEDC. The program as it presently stands could be interpreted as satisfying Criterion V, Instructions, Procedures, and Drawings of 10 CFR 50, Appendix B. However, the program is lacking in that it does not formally implement all of the requirements necessary to ensure that "as-constructed" satisfies "as-designed." This problem with the program may be attributable to one or more of the following:

a. Lack of an assessment of the adequacy of the scope and detailed requirements contained in the quality control program by an organization or organizations independent from the WBN CP or WBN CONST-QA organizations,

b. A misinterpretation of specific program requirements,

c. Design requirements not clearly and concisely defined

d. The seemingly continuous changing of requirements by EN DES or by NRC.

These problems were also found in the SQN program. Actions have been taken to correct these problems as stated in the preceeding paragraph. The adequacy of these actions cannot be determined at this time.

Failure to formally document closure sections in piping and to control preload in piping and nozzles as required in General Construction G-43 is an example of a significant design requirement that was not formally implemented in the quality control program for the installation and inspection of safety-related piping and supports. As a result, varying amounts of preload are suspected to exist in the piping and nozzles; because the 1/16-inch maximum misalignment of closure weld joints was not adhered to. External force of varying degrees was utilized to accomplish piping to piping and piping to nozzle alignments. Existence of this preload has the potential to invalidate the seismic analysis.

The WBN CP quality control, field change request, and nonconformance report programs as they presently stand have been identifying mislocated piping and hanger or support installation problems. These problems are resolved by way of QA audit reports, field change requests, and nonconformance reports. The preload problem had not been identified by the existing programs, and potentially could have gone undetected.

NSRS believes that the existing quality-control, nonconformance, and field change request programs coupled with the Phase I and Phase II programs to meet NRC IE Bulletin 79-14 (EDC 800827 020) could provide assurance that the final "as-constructed" configuration that evolves from the program satisfies the seismic design and analysis requirements. This NSRS belief is contingent upon:

- adequate training of inspectors in Phase I and II

- adequate definition of criteria to assess the results to the Phase II inspections and accept or reject the Phase I results

- adequate implemetation of both programs.

III. RECOMMENDATIONS

A. R-80-21-WBN-01 Preloading Problem

CEB with the assistance of the WBN CP should assess the magnitude and frequency of the existence of preload in the piping. Based on these findings CEB should specify corrective actions, if any, to eliminate the problem and to ensure the validity of the seismic analysis. In addition changes to the existing program that are acceptable to both organizations should be identified and implemented to prevent recurrence of the problem in piping being installed now and in the future.

B. R-80-21-WBN-02 Different Reference Points for Locating Piping and Supports

EN DES organizations that specify locations for piping and supports at WBN should review their current practices to determine the extent of this problem. Based on the findings of the review, these organizations should specify corrective actions to eliminate this problem on present and future drawings under their control and on drawings issued to the WBN CP or already implemented by the WBN CP.

C. R-80-21-WBN-03 Control of the Field Fabrication Sketch Program

The WBN CP should establish and maintain control of the field fabrication sketches for piping yet to be installed in WBN-1 or WBN-2. The goal of such an effort should be to not release a sketch for implementation until after it has been determined by CP engineering personnel to be consistent with the requirements on the design drawings.

IV. DETAILS

Two members of NSRS were onsite during the period December 10-12 to conduct a review of the WBN construction project (CP) program that governs the installation of safety-related piping and hangers. The NSRS personnel could not have asked for better cooperation from those persons contacted during the course of the review. Also, the positive attitude towards achieving a high degree of quality during construction activities expressed by all persons contacted is commendable.

WBN Safety-Related Piping and Supports Program

The review entailed interviewing persons at all levels of the WBN construction project organization to obtain viewpoints on the program that governs the installation and inspection of safety-related piping and supports. Specific areas covered in the interviews were the conceptualization, development, implementation, adjustment (refinement), problems encountered, suggestions for improvement of the program, and present status of program implementation.

The review revealed that the perspective of the program is consistent at all levels of construction project organization. The situation that currently exists with the installation of safety-related piping and supports may be described as follows:

- Ninety-two and seventy-seven percent of the safety-related piping has been installed in WBN-1 and WBN-2 respectively.

- Permanent hanger installation has progressed to thirty-eight and five percent complete respectively for units 1 and 2.

- Approximately twenty-five percent of the installed piping is determined to not be located per design when initial pipe location inspections are performed.

- This mislocated piping coupled with congestion caused by the installation of field routed (located) structures, systems, and components has made hanger installation per the original design difficult, if not impossible, to achieve.

- Varying amounts of residual stress due to the use of external force to effect weld joint alignment exists in the piping and equipment nozzles.

- The mislocated piping, the inability to install permanent hangers and supports per design, and the existence of residual stress in piping and nozzles taken individually or as combinations have the potential to invalidate the original seismic analysis for the plant safety-related piping systems.

The causal factors for this situation cannot be attributed to one single factor or organization. NSRS's understanding of the evolution of this situation obtained from the interview discussions is as follows:

- The WBN CP quality control program was formulated during the early stages of the construction project (approximately 1973). The program consisted of quality control instructions (QCI's) and quality control procedures (QCP's) which governed the installation and inspection of safety-related structures, systems, and components. The program formulation and background about the program are as follows:

    a. OEDC philosophy and practice in 1973 was for each CP to develop their own quality control program. In addition the OEDC philosophy on quality assurance was that the

Manager of EN DES and the Manager of CONST were responsible for organizing and directing their respective division's QA programs to attain quality objectives. The OEDC QA Manager was responsible for establishing basic QA Program policies and requirements, providing guidance, and overseeing the division's programs. Consequently, this resulted in each division developing their own QA Program with little regard for the other division's program except that provided by OEDC QA.

b. OEDC QA established policy and requirements for the QA programs in each division including a listing of all licensing commitments.

c. Both EN DES and CONST developed their respective QA programs which would govern the work within their organizations. For EN DES the Engineering Procedures (EP's) along with a number of implementing documents were developed. For CONST quality assurance and control program requirements were identified in the CONST QA Program manual.

d. EN DES produced over a time period drawings, specifications, General Construction Specifications, procurement specifications, memorandums, etc. all of which contained requirements or references to requirements that were to govern installation and inspection activities for safety-related structures, systems, or components.

e. All of this information was transmitted to the WBN CP for their use in constructing WBN.

f. Upon receipt of this information, the WBN CP sorted through all of this documentation to identify what activities required preparation of a QCI or QCP, and to identify all of the requirements that governed particular installation and inspection activities. From this the WBN CP developed the QC program (QCP's and QCI's) that would govern all safety-related activities. These QCI's and QCP's were reviewed and approved onsite by WBN CP personnel and WBN CONST-QA personnel. There was no requirement for any organization other than WBN CP and WBN CONST-QA to review the scope and detailed requirements for adequacy and consistency with licensing, program, and design requirements and/or commitments.

g. The WBN CONST-QA group at some point during the formulation of the QC program developed their QA audit program and audit schedule. The purpose of the WBN CONST-QA audit program was to provide some level of confidence that the QC program requirements were being adhered to during installation and QC inspection. The level of confidence to be provided was not formally quantified in any QA program requirements.

The program described in steps a. through g. was an acceptable way of developing and implementing the QC program. In fact, the program was believed to satisfy Criterion V, Instructions, Procedures, and Drawings of 10CFR50, Appendix B. However, two key steps, 1) clear and concise definition of licensing commitments, program requirements, and especially design requirements for use by the CP in developing their program, and 2) review of the QC program to assess the adequacy and consistency of scope and detailed requirements by some organization within OEDC other than the WBN CP, did not transpire. Consequently, the ability of the resultant QC and QA program requirements to ensure at some confidence level that the "as-constructed" configuration satisfied the assumptions and restrictions of the "as-designed" configuration was questionable.

Along these same lines, the WBN CONST-QA program and audit schedule was developed in conjunction with the QC program. This audit program was designed to provide confidence that the QC program requirements were being adhered to during the installation and inspection of safety-related structures, systems, and components. However, during this review it was learned that the audit scopes and frequencies were not statistically based. This needs to be resolved within the framework of the effective functioning of the QA/QC program.

- At WBN as at other TVA CP's, fitters prepare fabrication sketches from the design piping drawings. These fabrication sketches were and are required by the ASME code. The fitters at all TVA CP's have insisted that preparation of these sketches is their responsibility. Both the fitters and WBN CP personnel agree that the requirements on the design piping drawings have to be converted into field fabrication sketches for them to be conducive for field use.

  The original plan at WBN called for the fitters to prepare the sketches and for the engineering unit to review and approve each of these fabrication sketches prior to their release for actual work to begin. The review of the first series of fabrication sketches revealed that sketches contained errors in dimensions and component locations. However, due to a shortage of engineering unit personnel to review these sketches, the volume of sketches being produced by the fitters, and the large number of fitters whose work depended on the issuance of the sketches, the decision was made to waive the requirement for engineering unit review and approval of the sketches. Instead, the verification of piping location/ routing was to be made at the time the engineering unit oversaw the installation of permanent hangers and supports. Verification of location/routing of piping was not waived in this instance, rather it was only postponed to a later time frame in the life of the CP.

Later on in the life of the CP, the decision was made to set up a special hanger group separate from the piping group to oversee the installation of hangers and supports. Consequently, the piping location confirmation kept getting put off. The errors revealed in the early engineering unit review of fabrication sketches should have provided an early warning of potential problems in this area. This problem area needs resolution. (See recommendation C, Section III.)

- A problem was encountered in obtaining the permanent hangers and supports in a time frame consistent with the installation of the safety-related piping. Consequently, the permanent piping was installed on temporary hangers. These factors in and of themselves did not create a problem. However, installation of field routed (located) structures, systems, and components too close to the piping to allow installation of the permanent piping and supports transpired. This has and is causing difficulties with the installation of permanent supports.

- Use of different reference points to locate piping and to locate hangers resulted in further complications. Pipe location is referenced off of column centerlines while supports and hangers are referenced off of nominal wall, ceiling or floor faces. The result in some instances is a support or hanger that totally misses the piping that it is supposed to support, even though both pipe and support are installed per their respective design requirements. This problem with the reference points needs to be resolved. (See recommendation B, Section III.)

- The requirements contained in General Construction Specification G-43, "Support and Installation of Piping Systems in Category I Structures," are being formally implemented to varying degrees in the quality control program for the installation of safety-related piping, hangers, and supports. The reasons for this are:

  a. The purpose of G-43 is to establish minimum requirements for the support and installation of piping systems in Category I structures to assure that the piping is installed in such manner as to validate the analyses of the piping systems and insure conformance with the intended design of the system support scheme. This implies that the minimum requirements are all contained in this document. A closer examination reveals that G-43 itself requires knowledge of G-32, appropriate plant piping and support documents, manufacturers' recommended installation procedures, etc., to have a complete understanding of the requirements for the support and installation of piping systems. The situation just described emphasizes the point that the requirements governing the installation and inspection of safety-related piping and supports are not clearly and concisely defined in one or a relatively small number of documents.

b. The applicability of G-43 is to all piping, and piping supports installed in Category I structures with certain exceptions for embedded piping and piping provided as an integral part of prepackaged equipment provided by a vendor. G-43 continues by stating that procurement, materials, fabrication methods and details, inspection, and test requirements of the piping are not within the scope of G-43. This statement appears to conflict with the purpose of G-43 as stated in a. above; because if G-43 is to be the governing document to assure that the piping is installed in such a manner as to validate the analyses of the piping systems, then such areas as procurement, materials, fabrication methods and details, and especially inspection which have the potential to invalidate analyses must be controlled by some document. In light of this applicability philosophy, it is not surprising that G-43 reads as a general requirements document and does not contain a listing of the critical attributes with specific acceptance criteria that should govern the installation and inspection of safety-related piping and supports.

c. G-43 does contain some specific requirements such as the 1/16-inch maximum allowable misalignment of pipe joints while swinging free on supports for closure connection final assembly. This specific requirement is in G-43 to prevent significant preload on the final assembly connection which is a general assumption used in the seismic analysis. The concern for minimizing preload is justified per the seismic analysis; however, this requirement does not reflect conditions achievable from a constructability standpoint or criteria which are consistent with the ASME code on weld joint misalignment and on "cold-springing." Since the ASME code is utilized more extensively than G-43 and since the weld joint fit-up inspections are performed by welding unit personnel, the requirements of G-43 in this area tend to not be vigorously followed. Consequently, a documented program to minimize pre-load due to use of external force to effect closure connection alignment was never implemented at WBN. Piping fit-up was generally checked against the requirements of the ASME code. As a result, instances of preload outside the limits allowed by G-43 and subsequently the seismic analysis may exist in nozzles and piping at WBN. The impact of this situation on the seismic analysis cannot be determined until after the extent and magnitudes of the preload are determined. The most important point is that a situation exists in the field which may indeed invalidate the seismic analysis. Consequently, this situation must be evaluated and resolved. (See recommednation A, Section III.)

The varying degree of formal implementation of the requirements in G-43 has contributed to the problems being experienced with piping, hanger, and support installation. Also, the failure to have a formal program to minimize residual stress in piping and nozzles for closure welds can be attributable to the reasons stated above.

- The WBN CP as other TVA organizations has experienced difficulties with recruiting qualified or qualifiable personnel due to the competition (salary and benefits) for this type of person in the marketplace. This is evident in the results of recent WBN recruiting trips. In addition, the same competition makes it difficult for TVA to retain these people once they are hired. The problem of recruiting and retaining personnel when coupled with pressure to meet the construction schedule forced first line supervisors to adopt crisis management techniques. Consequently, supervisors did not have time to effectively train, utilize, and supervise the activities of their subordinates. This resulted in schedule and cost impacts due to the rework required to correct first time mistakes.

All of these circumstances when combined produce the situation that exists today with the installation of safety-related piping and supports and the seismic analyses for these piping and supports. Refer to figures 1.0 and 2.0 for cause and effect relationships as perceived by NSRS.

NSRS recognizes that the existing quality, field change request, "as-constructed," and nonconformance programs at WBN have been and would continue to formally identify mislocated piping and support installation problems for resolution by EN DES. These programs coupled with the Phase I and Phase II programs implemented after the IEB 79-14 review at SQN would provide assurance that the "as-constructed" configuration satisfied "as-designed" requirements. With regard to the preload or residual stress problem, the WBN CP had already recognized that the sequence of piping installation requirements in G-43 had not been implemented. Per reference 11 a meeting had been held with EN DES in an attempt to resolve this problem. The discussion in the meeting centered around documentation of closure sections and the cost to rework the piping systems to satisfy this requirement. Neither the significance and ramifications of preload nor the requirements to minimize preload in piping and nozzles, although discussed in the section of G-43 on piping installation sequencing, was sufficiently understood by the personnel interviewed until this review by NSRS. This understanding coupled with the knowledge of piping installation practices led to the conclusion that preload as discussed in G-43 does exist to varying degrees in safety-related piping and nozzles at WBN. Consequently, NSRS as stated in Section III of this report recommends that CEB determine the extent and significance of this situation and based on their findings propose actions to resolve this problem.

Comparison with NSRS Review Findings on SQN

NSRS in August 1980 issued a report entitled, "NSRS Assessment Sequoyah Nuclear Plant Compliance with NRC-OIE Bulletin 79-14 "Seismic Analysis for As-Built Safety Related Piping Systems"" (GNS 8C0814 001). This report and the two OEDC responses to the report were utilized by the reviewers in preparing for the review at WBN. The findings in this review at WBN support NSRS's supposition in the SQN report that " ... similar problems may exist or have the potential to exist with the adequacy of the seismic qualification of the "as-built" safety-related piping and hangers at all of TVA's nuclear plants."

V. Persons Contacted
WBN CP

T. B. Bucy - Supervisor, Hanger Engineering Unit
C. O. Christopher - Assistant Construction Engineer
F. H. Denton - Welding Inspector
J. Evers - Authorized Nuclear Inspector
M. A. Harper - Training Officer
L. J. Johnson - Supervisor, Mechanical Engineering Unit
B. S. Johnson, Jr. - Assistant Construction Engineer
J. M. Lamb - Supervisor, Mechanical Engineering Unit
A.L.B. Mayes - Steamfitter Superintendent
F. M. McGraw - Authorized Nuclear Inspector
R. W. Olson - Construction Engineer
A. S. Perry - Welding Inspector
A. W. Rogers - Supervisor, Quality Assurance Unit
F. Smith, Jr. - Supervisor, Office, Materials, and Civil Engineering Unit
J. B. Tubb - Assistant Electrical Superintendent
J. E. Wilkins - Construction Project Manager
S. Wolfe - Welding Engineer

VI. References

1. Division of Construction QA Manual

2. OEDC QA Program Requirements Manual

3. OEDC QA Manual for ASME Section III Nuclear Power Plant Components

4. TVA Design Specifications

   a. WBNP-DS-1935-2473-1

   b. WBNP-DS-1935-2618

   c. WBNP-DS-1935-2619

5. TVA General Construction Specification G-43, "Support and Installation of Piping Systems in Category I Structures"

6. WBN Construction Specifications

   a. N3G-881, "Identification of Structures, Systems, and Components Covered by the Watts Bar Nuclear Plant Quality Assurance Program"

   b. N3M-868, "Field Fabrication, Assembly, Examination, and Tests for Piping System for Watts Bar Nuclear Plant"

7. Division of Construction Quality Control Instructions

   a. WBNP-QCI 1.8

   b. WBNP-QCI 1.10

   c. WBNP-QCI 1.11

   d. WBNP-QCI 1.17

   e. WBNP-QCI 1.21

   f. WBNP-QCI 1.22

   g. WBNP-QIC 1.28

   h. WBNP-QCI 1.38

   i. WBNP-QCI 4.2

8. Division of Construction Quality Control Procedures

   a. WBNP-QCP 1.7

   b. WBNP-QCP 1.16

   c. WBNP-QCP 3.11

   d. WBNP-QCP 4.7

   e. WBNP-QCP 4.8

   f. WBNP-QCP 4.10

   g. WBNP-QCP 4.24

   h. WBNP-QCP 4.28

   i. WBNP-QCP 4.30

9. Watts Bar Nuclear Plant Field Instructions

   a. WBFI-G-7

   b. WBFI-M35

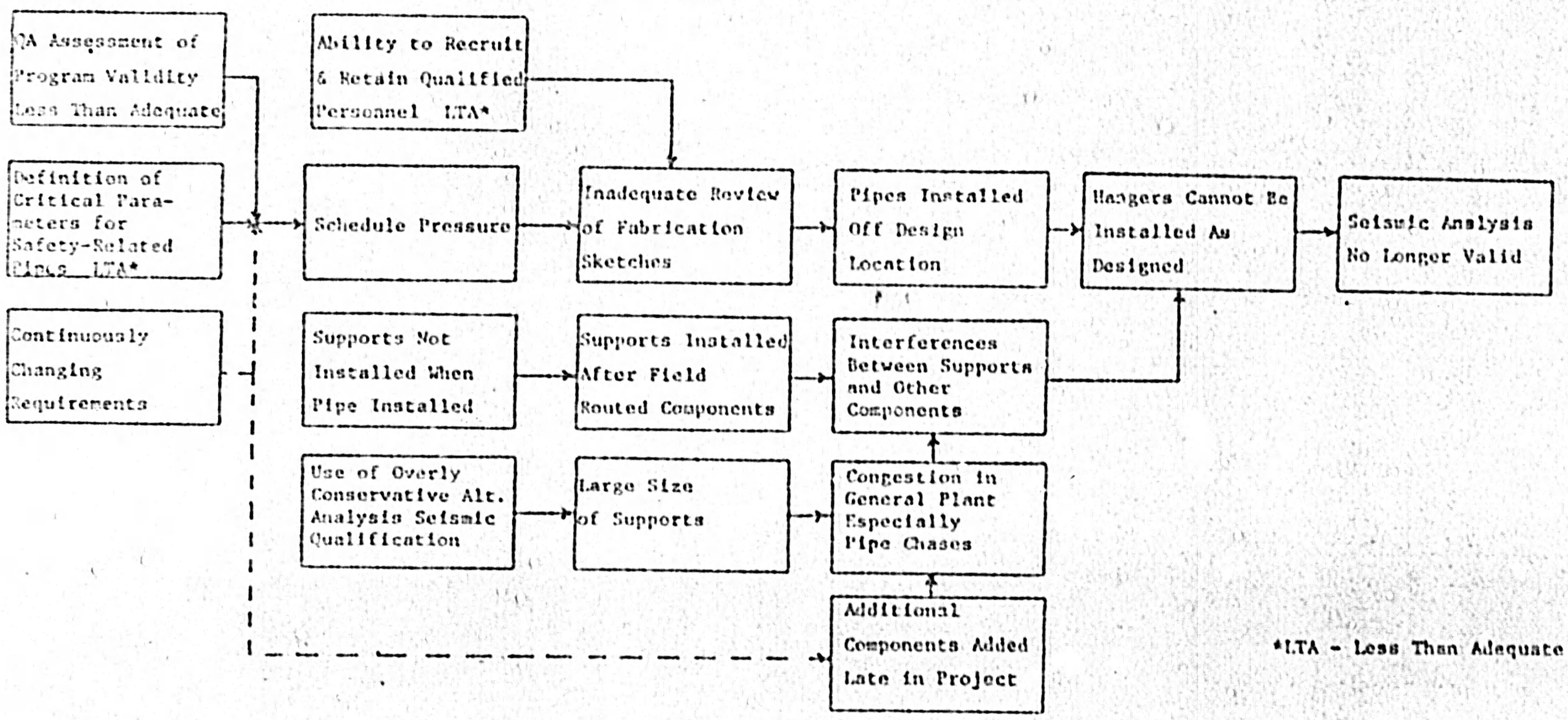
10. Memorandum from R. M. Pierce to J. E. Wilkins dated May 19, 1980 (MEB 800519 019)

11. Memorandum from J. E. Wilkins to R. W. Cantrell dated December 10, 1980 (WBN 801210 003)

12. 10CFR50, Appendix B

FIGURE 2.0

CAUSE/EFFECT CHART FOR
PIPE HANGER INSTALLATION
PROBLEMS



*LTA - Less Than Adequate

FIGURE 1.0

CAUSE/EFFECT CHART FOR
RESIDUAL PIPE STRESS