VI. DETAILS

A. COMMITTEE ACTIVITIES

The Browns Ferry (BFN) Technical Specifications required TVA to establish and maintain onsite and offsite review committees. ANSI N18.7 required the operators of nuclear power plants to establish an independent review program.

The onsite review committee requirements were satisifed by functions performed by the Plant Operations Review Committee (PORC). The PORC program consisted of Technical Specification 6.2.B and was implemented through the PORC charter and SP BF 1.10, "Plant Operations Review Committee."

The offsite review requirements were satisfied by the Nuclear Safety Review Board (NSRB). The NSRB program requirements were defined by Technical Specification 6.2.A and were implemented through the NSRB charter. A more detailed discussion of the programs and their implementation is provided under more specific headings in the following paragraphs.

1. NSRB

The NSRB was established to independently review various subjects relating to nuclear safety as required by technical specifications or as otherwise deemed desirable for safety reasons. The technical specification requirements were defined in the NSRB charter and appeared to be implemented from the quantative standpoint. However, NSRS found very little evidence that any desirable reviews beyond the Technical Specification requirements were being performed. The basis for the review activities was to keep TVA management and, particularly, the POWER Manager informed of the adequacy and success of implementation of the nuclear safety program. The following paragraphs provide a discussion of a number of weaknesses or potential weaknesses which NSRS believed could adversely affect the effectiveness of this independent review and management notification process.

a. The NSRB program had not been updated to require that the combined expertise of the NSRB include nondestructive testing as required by ANSI N18.7-1976. TVA has committed to the requirements of this standard.

b. The applicable version of ANSI N18.7 was issued in 1976 and indicated that the offsite standing review committee concept should be considered an interim approach and that owner organizations should make appropriate plans to effect an orderly transition to an organizational review concept to perform the

independent review function. In February 1978, the NRC endorsed ANSI N18.7-1976 through revision 2 of the Regulatory Guide 1.33. As part of the endorsement, the NRC listed certain sections of the standard that were sufficiently important to safety that the verb "should" would be considered as the verb "shall." These sections would then be enforced as requirements. The section of the standard that discussed the independent review functions was not included in the list of requirements. Therefore, the transition to an independent organization review process from the standing committee remains a recommendation.

TVA fits this expanding nuclear commitment concept, but there was no evidence that TVA had initiated any plans for the orderly transition specified in ANSI N18.7-1976 or that an evaluation had been performed to justify continuing with the standing committee approach. NSRS believes that this industry and NRC position should receive serious consideration.

The independent review concept utilized by TVA was primarily an after-the-fact review approach. In the cases where proposed activities were being reviewed, the safety evaluations were being performed by organizational reviews. As a committee to assure that the administration of safety activities are properly conducted, the NSRB appeared to be an effective, functional body. As an effective evaluator of nuclear safety it appeared to be deficient in timely information which is a key ingredient for any analysis. It is not clear that the NSRB was intended to function as a safety evaluation body and be involved in the resolution of current events. If this was the intent, more Board members should be selected from line organization positions that have direct access to operational information. If it was intended to perform an administrative function to assure that safety evaluations were performed by other organizations, then credit for independent evaluations should not be taken. The technical specifications can be satisified either way, but to satisfy the original safety intent, the independent review process should make enough independent evaluations to assure that the accepted system is functioning correctly and efficiently.

c. There was a possible conflict of interest built into the exisiting management structure controlling the NSRB members. The NSRB is in session only on a

part-time basis. The BFN NSRB charter required the Board to meet at least once per six months. In actual practice, the Board meets more frequently due to specially called meetings. However, it should be understood that participation in Board activities requires only a small portion of the workload of any Board member. During the periods that the Board was in session and functioning as a body, it was directly responsible to the Manager of Power. However, when the Board was not convened, each member reported to his "normal" supervisor and performed his routine functions. The NSRB was composed of a chairman and five members. The Supervisor of the Nuclear Safety Staff (NSS) served as Chairman of the NSRB when it was in session. This individual reported to the Manager of Nuclear Regulation and Safety (NRS) during the performance of his primary duties as Supervisor of the NSS. Another individual in the NSS was a member of the board. A member of the Regulatory Staff that reported directly to the Manager of NRS was a member. A representative from the QA&A Staff was a member of the Board. This individual's supervisor was the Supervisor of the QA&A Staff. The Supervisor of the QA&A Staff reported to the Manager of NRS. One representative from NUC PR and one from the Division of Fuels completed the NSRB membership. Therefore, when the Board was not in session, at least four of the six members of the Board were managed, directly or indirectly, by the Manager of Nuclear Regulation and Safety. Since a quorum of the NSRB could be satisfied by five members, four-fifths of the members could be managed by this one manager.

The NSRB charter states that no more than one-third of the members shall be from any one organization. However, at least two-thirds and possibly four-fifths of the members are from the NRS organization. The only apparent reason for limiting the number of members from each organization was to limit the influence of individual managers within the organization. However, it appeared, that a potential existed for the individual responsible for TVA licensing activities to strongly influence the decisions of the majority of the NSRB members.

It should be noted and clearly understood that no conditions were identified by NSRS during the review that would indicate or suggest that any attempt had been made by the Manager of Nuclear Regulation and Safety to exert pressure or to influence any

member of the Board in their decisions. The system appeared to be working primarily because of the individual strength and character of the incumbent manager. Nevertheless, NSRS believes that the system presented the potential for a conflict of interest

d. According to ANSI N18.7-1976, the purpose of the standing committee approach was to better utilize the available nuclear knowledge. This was originally accomplished by having the most knowledgeable engineers, supervisors, and managers also serve on the standing committee and make important reviews as a body based on discussions between individuals that had first-hand information and understood the information and its acutal or potential impact on plant safety and operations. The NSRB was not composed of this type people. The Board was made up primarily of staff personnel who received most information second hand and when the operating organization had time and the desire to provide it. If a decision is made to maintain the standing committee to perform the independent review function, a mechanism should be developed to assure that sufficient information is provided to the reviewers in a timely manner to support meaningful reviews. The timeliness and quality of information should be equivalent to that provided to the Director of NUC PR.

e. ANSI N18.7-1976 specifies the disciplines for which expertise must be maintained within the collective body of the NSRB. Eight specific disciplines are specified. Of these, NSRS concluded that the NSRB was weak in four. The four areas in which weaknesses were believed to exist were:

(1) Nuclear power plant operations

(2) Chemistry and radiochemistry

(3) Metallurgy

(4) Nondestructive testing

NSRS reviewed a list of TVA personnel that had been assigned to work with the NSRB as consultants. The personnel listed appeared to have the expertise necessary to satisfy the requirements of ANSI N18.7-1976 if effectively utilized.

2. **PORC**

ANSI N18.7-1976 does not require a standing committee approach for the review of plant activities. It holds that plant management is responsible for the review of plant activities as part of the normal supervisory responsibilities. Since the NRC has endorsed this standard, it would appear that the plant Technical Specifications could be amended such that PORC would no longer be required. However, the NSRS does not believe that this would be a prudent act. Our review indicated that the committee approach to the review of plant activities was both effective and practical. The committee members represented the most experienced engineers, supervisors, and managers at the plant. They appeared to be strong individuals willing to establish and defend their positions based on the best available information. A meaningful document review process had been developed and was being used. It would be difficult to conceive of a system that could better utilize the available resources in the review process. Information relative to all processes, systems, and events was readily available to the committee. No significant activity or event was likely to be overlooked by or concealed from the PORC.

Notwithstanding the overall high quality of performance of the PORC, one example of failure to comply with regulatory requirements was identified. A second item was identifed through the review of PORC activities which did not relate solely to PORC but represented a possible weakness in the assignment of responsibility in the area of unreviewed safety question determinations pursuant to 10CFR50.59. These items are discussed below.

a. It appeared that PORC was not performing a review of the adequacy of the QA program as required by Technical Specification 6.B.4.h. NSRS was unable to determine from PORC minutes that such a review was being performed. Members of PORC indicated that a formal review of the QA program was not being made. Most members felt that each time a procedure or an event was reviewed and corrective action was taken, the QA program was receiving a practical review and upgrading.

   Reviews and corrections of procedures and events may serve as a practical tool for upgrading the functional QA program at the site. However, these reviews are required by other Technical Specifications and it appears that the Technical Specification writers did not consider this adequate review of the QA program since they added the requirement for this review also.

NSRS understands that no frequency of the review was specified. If literally interpreted, the specification could be satisfied by a one-time review, by a continuous review and documentation, or by specific periodic review conditions to indicate problems or weaknesses in the program. NSRS believes that a structured plan for satisfying the Technical Specification requirement should be developed and implemented.

b. 10CFR50.59 states that, "The holder of a license authorizing operations of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test, or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question." The license holder is required to maintain records of changes to the facility, changes to procedures, and tests and experiments performed. The records must include a written safety evaluation which provides the basis for the determination that the change, test, or experiment does not constitute an unreviewed safety question.

NSRS understands that EN DES makes the unreviewed safety question determination for changes to the facility (hardware modifications). It was not clear during the review that responsibility for making unreviewed safety question determinations for changes to procedures described in safety analysis reports (SAR's) and for tests and experiments had been assigned within POWER or that such determinations were being made. This statement is based on discussions with POWER personnel and a review of the program for PORC activities. PORC members stated that PORC did not make unreviewed safety question determinations and except for plant modifications, they did not know who did. Some members also stated that no individual or group at the plant made unreviewed safety question determinations. A number of individuals in the central office in Chattanooga were of the opinion that PORC made the determinations when it was necessary for them to be made within POWER. The N-OQAM Part II, section 4.6 stated that, "Proposed safety-related STEAR's referred to OEDC shall be given a safety review by EN DES. This safety review shall determine whether an unreviewed safety question (as defined in 10CFR50.59) is involved." The N-OQAM did not require all safety-related STEAR's to

be referred to OEDC. SP BF 17.1, "Special Tests, Experiments, or Special Activities," established the method by which special tests, experiments, and other activities were handled. This SP did not assign the responsibility for making unreviewed safety question determinations. The SP required that form BF-12 be used as the cover sheet for safety-related STEAR's. This form provided a signature space for EN DES review confirmation. Form BF-13, "Unreviewed Safety Question Determination," was required by SP BF 17.1 to be completed and attached only for nonsafety-related STEAR's. This would seem to indicate that EN DES did unreviewed safety question determinations for all safety-related STEAR's. However, NSRS understands that NUC PR has the perrogative to perform the safety evaluations and unreviewed safety question determinations. One example of the exercising of this option is the lifting of the electrical leads to the cooling tower lift pumps last summer. At the exit meeting in Chattanooga on March 3, 1981, a NUC PR management representative stated that for situations requiring unreviewed safety question determinations by POWER, the determinations were made at the site. NSRS understood another management representative at the exit meeting to say that 10CFR50.59 did not apply to operational procedures at BFN and that this had been agreed to by the NRC. NSRS understood the POWER position to be that changes to the operational procedures prescribed by Regulatory Guide 1.33 and which implement the QA program at BFN did not require an unreviewed safety question determination in accordance with 10CFR50.59. If this is a correct understanding, NSRS wishes to review the NRC agreement or other basis for the conclusion. NSRS is not questioning any specific change, test, or experiment that has been accomplished. It is the management control system for assuring that the requirements of 10CFR50.59 are satisified that is in question.

It was interesting to note that the Technical Specifications did not require PORC to make the unreviewed safety question determinations. PORC was required only to review proposed changes to equipment or systems having safety significance, or which may constitute an unreviewed safety question pursuant to 10CFR50.59. This serves only as a check that EN DES is performing its function for plant modifications.

B. CORRECTIVE ACTION

Criterion XVI of Appendix B to 10CFR50 requires that measures be established to identify, evaluate, correct, and report significant

conditions adverse to quality, such as nonconforming materials, parts or components, failures, malfunctions, deficiencies, and deviations from regulatory codes, standards, and drawing requirements. TVA had satisfied these requirements by development of various written programs and mechanisms to ensure management that deficiencies identified both internally and externally to TVA were properly identified, evaluated, corrected, and reported. Following is a discussion of each program.

1. Externally Identified Deficiencies

a. Nuclear Plant Operating Experience Reports

TVA had implemented a program for obtaining Nuclear Operating Experience Reports from the nuclear industry, evaluating the experience and for disseminating it to personnel which must be cognizant of the event. The program was described in DPM No. N72A39, April 9, 1980, which established a review panel to perform evaluations to determine applicability to TVA's operating nuclear plants. NSRS verified that the mechanisms established by DPM N72A39 were in effect with the exception that the Nuclear Experience Review Panel (NERP) had been dissolved. The reviews were being performed and documented by line personnel assigned by the CO. This was in addition to the evaluations conducted onsite by the Shift Technical Advisor (STA) dissemination at the plant level. At the time of the review, NUC PR had drafted a new revision to the DPM which would eliminate the NERP panel and place the responsibility for assigning the evaluation of the event and tracking its resolution to the Reactor Engineering Branch at the CO.

Notwithstanding the changes that are taking place, NUC PR appeared to have an adequate system to assure that events that occur at other plants were evaluated for applicability to TVA plants and corrective action was taken and documented.

b. Vendor Information Letters

The CO had established mechanisms to assure that responsibility had been assigned to perform and document an evaluation for applicability, to take corrective action if necessary, and to track the evaluation and corrective action to completion for General Electric (GE) Service Instructions Letters (SIL's) and various Maintenance and Technical Instruction Letters. DPM BF72A2, August 28, 1973, placed the responsibility on

the plant manager for conforming to applicable GE SIL's. In view of the mechanisms established at the CO, this DPM appeared to be outdated.

A review of the CO records and discussions with CO personnel indicated that applicability reviews were completed and that the corrective action taken for all of the issued GE SIL's had been documented.

c. NRC-OIE Bulletins, Circulars, Information Notices, Inspection Reports and Special Requests for Information

The CO had established mechanisms for evaluating, correcting, documenting, and tracking of NRC IEB's, IEC's, IEIN's, NRC inspection reports, and special requests for information. These mechanisms appeared to be working satisfactory, however, they had not been described in written form. NUC PR personnel stated that the plans were to centralize coordination of these type items in the Reactor Engineering Branch at the CO.

BFN management had established an administrative procedure, SP BF 21.14, "Response to NRC Bulletins, Circulars, Information Notices, and Other Requests for Information from NRC and Other Regulatory and Inspection Agencies," dated December 30, 1980, assigning responsibility for review, coordination, response, corrective action, and tracking. The site compliance supervisor had been assigned the responsibility of establishing and maintaining a commitment control tracking system for these type items. This program was being developed at the time of the review.

In addition, SP BF 15.17, August 27, 1979, had been developed to provide guidance for preparing responses to NRC inspection reports and OPQA&A audit reports. The OP and NUC PR had recognized weaknesses in the programs for controlling commitments made to outside agencies. As a result, they have been in the process of developing tracking or status systems to assure management that commitments would not be overlooked.

The OP commitment list consisted of commitments made as a result of NRC bulletins and inspection reports, NRC requests for information, special audits required by the Technical Specifications, special reports due, various miscellaneous requests, and commitments made in response to NUREG-0737. The NUC PR list consisted

of approximately the same type items; however, commitments to the TVA Blue Book were included.

Discussions with OP and NUC PR personnel indicated that these lists were in the process of being developed and did not contain all of the commitments. NUC PR representatives stated that it was their intent to develop a commitment list for each plant which could be utilized by the site compliance supervisor such that dual lists would not be required.

DPM N76A9, July 12, 1976, "Commitments to Outside Agencies," established a commitment control program which appeared to NSRS to be out of date. However, the site had recenlty issued SP BF 3.8 which assigned the responsibility for maintaining status of these commitments to the site QA Supervisor. Discussions with the site QA Supervisor indicated that responsiblities for tracking and maintaining status of all commitments to outside agencies were recently shifted to the compliance supervisor.

In addition, DPM N77A11, October 4, 1977, "Open-Item Status - Followup System," was established to give the plant manager the responsibility to establish programs for followup of outstanding commitments, requirements, and open items for both internally and externally identified items.

In summary, mechanisms had been established at the office and divisions levels for followup of externally identified items, however, these mechanisms were not well defined in administrative controls, particularly at the office and division levels. NUC PR was in the process of centralizing the programs for ensuring that externally identified items properly evaluated, responded to, corrected, and documented such that the Reactor Engineering Branch would provide the overview for all identified outside of NUC PR. A program for followup of commitments described in DPM N76A9, July 12, 1976, was not being implemented, however, the OP and NUC PR had recognized the need to develop a new commitment control program and were in the process of doing so.

Site procedures at the site established responsibility for evaluating, correcting, responding to, documenting, and tracking of internally identified items, however, it appeared to NSRS that SP BF 3.8, December 11, 1980, was not needed because the responsibilities stated in it were assigned to the Compliance Supervisor in BF SP 21.14.

2. Internally Identified Deficiencies

The programs established at BFN to meet the requirements of N-OQAM, part II, sections 7.1, 7.2, and 7.3, of identifing, evaluating, correcting, and reporting conditions adverse to quality identified by plant employees were contained in the BF SP. The overall program was described in BF SP 10.1, "Correction Action Policy," while specific programs were described in subsequent SP's. Following is a discussion of the specific programs.

a. Licensee Reportable Event (LER) Determination

SP BF 10.2, "LER Determination," described a program for identifying, evaluating, correcting, and reporting events which were reportable to the NRC. Additional requirements were contained in SP BF 12.14, "Policy for Adherence to TS Limiting Conditons for Operations and Reporting of Significant Events." SP BF 15.1, "Implementation of Reporting Requirements for Nuclear Plants," SP BF 15.2, "Operational Event Report," DPM N74A8, specified the extent of CO review, and DPM 73015 specified what items were reportable.

NSRS reviewed the above described program and implementation to assure that internally identified deficiencies were identified, reviewed by supervision, evaluated for corrective action, corrected, tracked, and reported to the NRC as required. No problems were identified in the implementation of the program; however, the site program did not provide for a 10CFR21 evaluation. The evaluations were being completed based on instructions from DPM 77A14 and DPM N74A8, September 28, 1979, and were being directly implemented at BFN.

b. Corrective Action Reports

SP BF 10.3, "Noncompliance Determination," provided a mechanism for identifying, documenting, correcting, tracking, and reporting events or conditions adverse to quality other than those that were reportable under the TS (LER's), radiological incidents, and security degradations.

NSRS reviewed this program and its implementation and concluded that the implementation of the program was not as effective as it could be (paragraph K.1.a). It was evident that this program had received increased management attention during the past year by addition of a computerized tracking system.

c. Other Programs

The Browns Ferry Standard Practices also provided programs for the identification of and followup of radiological incidents and security degradations. Although these programs were reviewed, no attempt was made to review the implementation of these programs.

d. Plant Tracking Programs

Various tracking programs had been established at plants to assure that actions were taken in a timely manner and were not overlooked. Generally, the Compliance Supervisor was given responsibility to track the status of items identified by groups external to the plant such as OPQA&AS or NRC.

The QA Supervisor was responsible for tracking the status of corrective action reports (CAR's). In addition, Plant Services Supervisor had developed tracking programs for a number of items, such as LER's, security degradations, and radiological incidents. NRC violations and open items and IE bulletins and circulars were also being tracked by Plant Services, and the CO was providing tracking of various commitments to outside agencies.

In summary, it was apparent that duplication existed in some of the tracking programs and that program changes were being made at both the CO and the plant.

e. Trending Programs

The Plant Manager had devised a program for trending various performance indicators for adverse trends. Included in the program were graphical and statistical representation of unit scrams, NRC violations, LER's caused by personnel errors, security degradation determinations, radiological incident reports, and CAR's.

The Plant Manager stated the program was still under development and was not described in administrative procedures. At the time of the, the Plant Services Section was developing the trending reports for the Plant Manager on a monthly basis.

NSRS was impressed with this program, and the Plant Manager should be commended for his efforts in this area.

3. OPQA&A Staff Audits of Corrective Action

BFN Technical Specification dated February 1980 established a requirement for NSRB to audit the corrective action programs at BFN at least once each six months. A review of the audit reports issued in 1979 and 1980 indicated the following audits relating to corrective action were in the files:

Audit Report No.	Date of Audit	Scope of Audit
BF 78-12	12/4-8/78	System for and status of CAR's to include followup by QA, effectiveness of QA, and retention of CAR's
BF 79-09	12/1012/79	Plant SP's BF 10.1-10.6, CAR system to include recurrence control

It appeared to NSRS that the audits performed to meet the TS requirements were not adequately scoped to give management assurance that the corrective action system was working effectively because all deficiencies affecting quality were not identified, evaluated, corrected, and reported by use of the CAR system. For instance, the audits did not include the LER program which handled the identifying, evaluating, correcting, and reporting of items identified in the N-OQAM as "significant conditions adverse to quality." In addition, audits had not been conducted of the various programs for handling externally identified deficiencies.

C. EMERGENCY PLANNING

The Division of Occupational Health and Safety (OC H&S) through its Radiological Hygiene Branch (RHB) was responsible for the overall coordination and implementation of TVA's Radiological Emergency Plan (REP). Individual divisons EN DES, NUC PR, OC H&S, and each nuclear plant were responsible for preparing their own implementing procedures consistent with the REP.

With the exception of contingency plans to provide for the replacement of on-duty site personnel during prolonged emergency conditions, emergency planning programs appeared to have been developed and implemented that met or exceeded regulatory requirements and TVA commitments. Plans and preparations for drills and exercises to demonstrate operability of the emergency plan were being made. The preparations included monitoring of the exercises and followup action to verify proper corrective action when appropriate.

Discussions were held with the RHB Radiological Emergency Planning and Preparedness Group (REPP) regarding their role in the overall REP. They stated the development of Implementing Procedures (IP's) for BFN was their number-one priority and would be complete on February 23, 1981. Personnel at BFN indicated that the February 23 date was a little optimistic. The REPP had developed a tracking system for all NRC commitments to the REP. Once reviewed by NUC PR and EN DES this system should be operational. The REPP is keeping a training record file as required in the REP. They have not performed any cross-checks to assure themselves that personnel requiring training by the REP have in fact received it. According to REPP they plan to do this once the BFN REP and IP's are complete.

The Muscle Shoals Emergency Control Center (MSECC) has been rearranged since it was determined, during the June 1980 Sequoyah emergency drill, to be too congested. The new layout appeared to be effective and should reduce the congestion. Each of the subgroups in the MSECC (offsite dose assessment, etc.) has its own office area, where as previously the subgroups were essentially all together. The duty roster for emergency personnel was being prepared on a monthly basis. Each person on the roster was assigned a pager and was restricted to 15 miles from the MSECC while on duty. Training of selected emergency personnel is conducted on a weekly basis. RHB was responsible for developing scenarios for emergency drills and exercises for auditing TVA's performance. Three exercises are tentatively scheduled for the next 12 months. Each will test the total TVA REP and will include State participation. The tentative schedule includes SQN in July 1981, BFN in September 1981, and WBN in March 1982.

The NUC PR offsite Central Emergency Control Center (CECC) in Chattanooga has been rearranged and increased in size. This change was determined necessary after the June 1980 SQN drill. The facility included 1,500 square feet of space for the CECC staff, 700 square feet for the duty specialist, 300 square feet for a conference room, 2,100 square feet for NRC office space, and 3,000 square feet for media briefings. This arrangement appeared far superior to the previous one and should enhance operations. Emergency personnel were assigned on a monthly basis and were provided pagers. A duty specialist was present 24 hours per day and had a visible status board with the emergency personnel listed along with their telephone and pager numbers. Training was being conducted weekly for selected emergency personnel. This, like the MSECC training, was far in excess of what was required by the REP was but considered by NSRS to be appropriate until all emergency personnel have been trained. It is highly probable that the six-month minimum training frequency required by the REP will never be a satisfactory or realistic timeframe and that training will probably be conducted more frequently. Special training was provided to BFN personnel by a NUC PR staff member

on the REP. This training was highly praised and well accepted by the site personnel involved.

At the time of this appraisal, BFN was in the process of developing IP's for the new REP. It was considered inappropriate to evaluate the site in this area.

A new CRT data transmission system was being installed. At the time of this review all three Emergency Control Centers were linked together. Once complete this system will link each offsite emergency center and the states of Alabama and Tennessee together, allowing hard-copy data transmission between each one. This is considered a significant improvement over the emergency communications network already in place.

The overall TVA policy of decentralized offsite emergency support centers versus a near-site emergency support center was evaluated. Considering the large number of nuclear facilities coming on line in the future and the large resource of available expert personnel located in Knoxville, Chattanooga, and Muscle Shoals, the benefit of this decentralized approach appears to far out weigh the benefit of having a fewer number of experts nearby.

The effort and support observed in the development of the REP by both RHB and NUC PR appear exemplary. A great deal of effort will continue to be required to keep the plan and personnel up to date and in a state of readiness.

One area deserving review involves the replacement of on-duty. onsite personnel during a prolonged emergency. Various factors that should be included in the review is the possibility of an offsite evacuation which involves off-duty personnel and hampers normal access to the plant. The development of plans and instructions, assembly points, and alternate methods of access to the plant should be explored. This scenario was not addressed in the REP or IP's.

D. ENVIRONMENTAL MONITORING

The responsibility for environmental monitoring assessment was assigned to the Radiological Assessment Group within RHB. Analysis of environmental samples (water, fish, milk, vegetation, etc.) was performed by the OC H&S Laboratory Services Branch and data was provided to the Radiological Assessment Group. Environmental TLD monitoring was performed by the RHB Radiation Exposure Management Groups, and data was provided to the Radiological Assessment Group. Offsite discharges of gaseous and liquid radioactive waste was monitored by the BFN Results Section and data was provided to the Radiological Assessment Group. Adequate programs in the area of environmental monitoring had been developed and implemented to satisfy regulatory requirements and TVA commitments.

On the priority list of things to do for the Radiological Assessment Group was an evaluation of the offsite sample locations with respect to current meterological conditions, sample collection methods, and plant discharges. Also included in this evaluation was an attempt to correlate recorded plant discharges with measured environmental samples. Because of other priorities this study had not substantially progressed, and attempts to correlate discharge data and environmental data had produced no correlation. The lack of correlation could be due to the relatively low levels predicted and measured in the environment, the method of determining the amount of radioactive material discharged, sample locations, or any number of other limiting factors. This lack of correlation adds support to the perception of the need to perform the desired study; however, this review was not detailed enough in this area to determine the level of effort that should be devoted to this study. However, the RHB Quality Assurance/ALARA Staff in audit report RHB/QA-80-1 recommended this study be undertaken and periodically reviewed for adequacy.

There has been a problem with the lack of documentation of the logic and factors used in developing computer programs that calculate environmental effects of radioactive discharges. The RHB Quality Assurance and ALARA Staff documented this problem in their audit report RHB/QA-80-1. At the time of this review the Radiological Assessment Group was working on this documentation and had completed the one for ground water, were finishing the one for river water, and were starting the one for gas.

Based upon this appraisal it appears that the environmental monitoring program is receiving adequate management, and the interfaces between responsible participating groups appear to be well defined and functioning smoothly. The technical aspects of the program were not evaluated due to time constraints.

E. FIRE PREVENTION/PROTECTION

The evaluation of the fire prevention/protection program included an assessment of the program establishment, adequacy, and implementation to ensure control of fire hazards and maintenance of fire protection equipment. A program had been developed and was being implemented to provide periodic surveillance for the existing fire protection systems and to maintain them in an operational condition. A concentrated effort did not appear to be directed toward the identification and correction of deficiencies or toward a systematic upgrading of existing systems. A considerable effort was being expended in the correction of specific deficiencies which had been identified by outside organizations such as NRC and TVA contractors. NSRS understands the near-term manpower constraints that were imposed by the significant reactive effort resulted from the large number of findings by the outside audits. However, we believe that the future plans of NUC PP

should include a routine program to review and update the fire prevention/protection systems. The review in this area did not include an indepth examination of physical equipment.

F. INSERVICE INSPECTION PROGRAM

The Director of Nuclear Power had been assigned responsibility for assuring quality in inservice inspection and testing activities at operating nuclear plants. This responsibility had been delegated functionally to the Nuclear Central Office (NCO) Codes and Standards Section, NCO Baseline and Inspection Inservice (ISI) Section, and plant operating staff. Separate programs existed for nondestructive examinations, system pressure tests, and pump and valve performance tests. The plant staff had been given the lead for developing and implementing programs and procedures for system pressure tests and also pump and valve performance tests. The Codes and Standards Section provided technical support to the plant staff, developed the ISI (NDE) program which was implemented by the Baseline and ISI Section, and coordinated preparation and submission of all program plans to the NRC for approval. The Metallurgy and Standards Group (Controls and Test Branch), which contained the Codes and Standards Section, provided other engineering expertise as needed. This group was also responsible for coordinating the upgrading of the maintenance and modification programs (see VI.H, "Maintenance") of NUC PR to satisfy requirements of section XI of the ASME Boiler and Pressure Vessel Code.

The ISI program must satisfy the requirements of both the Technical Specifications and 10CFR50.55a(g), "Inservice Inspection Requirements," which incorporates by reference section XI of the ASME Code. There was additionally a body of regulatory correspondence containing additional requirements and clarifications that must be satisfied.

The policies, directives, organizational documents, and administrative procedures that established the ISI Program for BFN were reviewed. These documents appeared generally to fulfill the regulatory requirements listed above. Some observations were made as follows:

1. ISI activities had been conducted in accordance with Technical Specifications and program plans that had not received NRC approval. The record showed that TVA had been unable to obtain NRC action on these matters. TVA had met all requirements for preparation and implementation of programs to the extent possible. To facilitate timeliness in future NRC reviews, TVA had proposed that the start dates for future ISI program intervals be synchronized so that a single review of the program plans could be made by NRC to satisfy the requirements for all three BFN units.

2. NUC PR had commenced work to upgrade its modification and repair program, as well as ISI, to satisfy fully the requirements of section XI of the ASME Codes. (See discussion under VI.H.) This would principally require independent verification by an authorized nuclear inservice inspection (ANII) that section XI requirements were being satisfied. Program and procedural changes to provide for ANII verification of ISI data were being prepared.

3. In August 1980 NUC PR reported that ISI testing requirements as specified in section 4.6.G.6 of the BFN-1 Technical Specifications, had not been performed during the successive refueling outages. To prevent recurrence NUC PR had determined that augmented ISI requirements would in the future be incorporated into program plans by appendix to control the required activity.

   The inspector reviewed documents related to ISI activities performed during the refueling outage for BFN unit 2, cycle 3, to verify the N-OQAM requirements were being satisfied:

   a. An ISI scan plan had been developed and approved by authorized personnel.

   b. NDE inspections had been performed by properly certified personnel.

   c. Evaluations of a flow indication had been performed by certified personnel.

   d. A completed report had been prepared in accordance with section XI of the ASME Code and had been submitted to the NRC.

   Program awareness of NUC PR personnel was examined during discussions with NUC PR's supervisory personnel both onsite and in the nuclear central office. It was concluded that these personnel had generally a clear understanding of responsibilities of their position and an awareness of the resources and capabilities of their own sections and those with which they interfaced.

G. LICENSED ACTIVITIES

The review of this area included management control programs over the organization, administrative controls, record and document control, plant operations, information flow, and surveillance testing.

The POWER policy for these areas was contained in TVA Topical Report (TVA-TR-75-1A) on QA for Design, Construction, and Operation of Nuclear Power Plants. For these areas POWER had committed to the

requirements of ANSI N18.7-1976, Regulatory Guide 1.33-February 1978, and the conditions of the license and ANSI N45.2.9-1974. In addition, special commitments have been made as a result of lessons learned subsequent to the TMI-2 incident.

It appeared to NSRS that commitment to the above described standards for BFN is beyond those required by the NRC for an average plant in the industry of the same vintage.

The policy was implemented at the NUC PR level by the Nuclear Operational Quality Assurance Manual (N-OQAM) and the DPM. The requirements delineated in the DPM and N-OQAM were implemented at BFN by the BF SP's. Additionally, Section Instruction Letters (SIL's) had been established at BFN to amplify guidance in the BFN SP's.

In addition to the above described administrative control program, NUC PR was developing a goals and objectives program to assist management in emphasizing plant performance, plant upgrade, and budget. The implementation plan was in the process of being developed to implement the goals and objectives.

Feedback programs were also being developed to provide management information regarding plant and regulatory performance. A personnel accountability program that related personnel performance with plant and regulatory performance was being studied at the plant level but was not being implemented.

1. <u>Organization and Staffing</u>

   a. An organization had been established by NUC PR to assure effective operations of BFN with management overview and technical support. Responsibility statements had been developed and staffing levels appeared appropriate. NUC PR management was aware of and attempting to correct several organizational weaknesses as follows:

      (1) The organization (both at the plant and the division levels) was in a transitional period due to reorganizations which had taken place. As a result of these reorganization changes in programs, changes in responsibilities were taking place; therefore, functional responsibility statements did not reflect, in all cases, current responsibilities. However, NSRS believes these will be corrected by current actions that are under way.

      (2) NUC PR management had encountered a problem of maintaining qualified personnel at the central office and at BFN. This was particularly true for the positions of Assistant Unit Operator and Unit Operator. Apparently personnel in these classifications were leaving TVA

for nonoperating positions. NUC PR management stated that the reason for the high turnover rate in these classifications was the imposition of more stringent NRC qualification requirements, low pay, and the scrutiny the operators were under when mistakes were made.

NUC PR management had performed salary comparison studies both internal and external to TVA and had increased nuclear license differential pay which may aid in the resolution of the problem. NSRS believes that all causal factors may not be under the control of NUC PR, however, it did appear that more could be done in the area of career planning and indoctrinating personnel regarding the need for management overview, independent audit, and personnel accountability in the operation of nuclear power plants.

b. During the review NSRS observed several deficiencies which NUC PR management was aware of that appeared to be contrary to the BFN Technical Specifications or the TVA Topical Report on QA as follows:

(1) BFN functional organization was different from the organization described in the Technical Specifications in that an Assistant Plant Manager had been added to provide supervision over the operations and engineering activities.

(2) The Division Operations Support Staff had been dissolved, and two of the responsibilities described in the TVA Topical Report on QA had been transferred to the NUC PR Reactor Engineering Branch (REB).

(3) The Nuclear Experience Review Panel, as described in the TVA Topical Report, had been dissolved and responsibilities were being assumed by the line management personnel. (See section IV.B for additional details.)

Subsequent to the review, NSRS reviewed a proposed Technical Specification change draft which would resolve the deficiency indicated in item (1) above.

2. <u>Administrative Controls</u>

The reviewer verified that the administrative controls had been developed and were functioning both at the central office and at the plant. The division level procedures (DPM's and N-OQAM) did not implement all the requirements of the Topical Report and a number of the DPM's were out of date. The DPM's (15 volumes) were assigned such that

they were difficult to use. NUC PR had recognized the need to rewrite the N-OQAM such that it would implement the requirements of the QA Topical Report and they had started an effort to reorganize and reduce the volume of the DPM's so that they would not contain extraneous information and would provide sufficient information to ensure uniform programs at the nuclear plants. One example of where the N-OQAM did not implement the QA topical report is as follows:

a. The N-OQAM did not address the plant QA survey program.

b. Several of the DPM's that appeared to be out of date are as follows:

(1) N72A36, May 26, 1978

(2) BF76M12, January 7, 1979

(3) N72A39, April 9, 1980

(4) BF72A2, August 28, 1973

(5) N76A9, July 12, 1976

The BFN SP's were generally up to date and adequate to assure that requirements were met. In cases where guidance was not provided in division documents, the SP's directly implemented the requirements in source documents (TVA QA Topical Report). Only minor discrepancies were noted in the BFN SP's primarily due to recent responsibility changes due to organizational additions. BFN SP's were divided into what appeared to be logical categories; however, it was necessary to review a number of SP's to determine whether all the requirements which pertain to a particular functional area, such as maintenance, had been accounted for. For this reason, the SP's appeared more complicated than they needed to be. As an aid to plant supervisors, the responsibilities for performing various functions described in the SP's had been placed on the plant computer. This enabled responsibility sorts to be made for individual supervisors. In addition, a SP cross-reference was devised as an additional aid to personnel using and revising the SP's.

3. <u>Record and Document Control</u>

TVA QA Topical Report committed NUC PR to establishing a program in conformanced with Regulatory Guide 1.88 (revision 2), October 1976, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records." Regulatory Guide 1.88 endorses ANSI N45.2.9. TVA took exception

to Section 5.6 of ANSI N45.2.9 in that active records may be stored in one-hour fire-rated file cabinets as opposed to the four-hour fire-rated file cabinets recommended in the standard.

N-OQAM, part III, section 4.1 stated that active QA records may be temporarily stored in cabinets near the work location for periods up to three months. Certain records may be stored for longer than three-month periods; however, these should be specified in the N-OQAM.

NSRS reviewed the record and document control program at BFN and identified the following weaknesses which NUC PR management is aware of and in the process of correcting:

The existing program described in the N-OQAM, Topical Report and BFN SP's regarding plant records was in the process of being implemented. Full implementation had not been achieved as follows:

a. A large backlog of records existed in the permanent record storage vault to be microfilmed.

b. Records were kept in an active status in section file cabinets for periods greater than three months. This was allowed if it was specifically in the N-OQAM; however, in this case it had not been.

In addition, NSRS was concerned that protection of records was not provided during the period (approximately seven days) that they were awaiting microfilming in the Document Control Center.

4. <u>Plant Operations</u>

The review in this area included control over operating evolutions, equipment control, shift turnover and equipment status, evaluation of transients, QA of operational activities, operations responsibility, and documentation of operational activities. Following are the details of the review.

a. Plant operating evolutions were controlled by detailed written procedures. BFN management had developed a program which is in conformance with the requirements established in the TVA QA Topical Report and the Technical Specifications. Technical reviews of the Operating Instruction (OI's) and the Emergency Operating Instructions (EOI's) were not performed, however, the program was reviewed to verify conformance with Regulatory Guide 1.33. Several discrepancies were identified as follows:

(1) General Operations Instruction (GOI) 100-1 on Integrated Plant Operations contained a statement as follows: "This instruction is intended as a general guideline for plant operations and does not, nor is able to, cover all possible operating conditions. Strict adherence is not required and deviation is permissible, with the consent, and under the direction of the shift engineer." NSRS contends that this statement gives the Shift Engineer authority to make temporary changes not intended by ANSI N18.7-1976, section 5.2.2 or BFN Technical Specification 6.3.B which limit his temporary change authority to "non-intent" changes only, except in the case of an "emergency."

(2) Several instructions recommended in Regulatory Guide 1.33, February 1978, as emergency instructions were in the abnormal section of the OI's. These are as follows:

(a) Loss of Service Water - OI 25, 26

(b) Loss of Protective Channel - OI 94

(c) Conditions Requiring use of Standby Liquid Control System - OI 63

(d) Intrusion of Demineralizer Resin in Primary System - OI 69

(e) Recovery from Reactor Trip - GOI 100

Two of the above procedures, (c) and (e), were listed in the N-OQAM, part II, section 1, November 10, 1980, as examples of potential emergencies. The primary difference between the emergency and abnormal section OI's was that the operator is required to commit to memory the immediate action steps of the emergency procedures as specified in N-OQAM, part II, section 1.1, paragraph 3.0.

(3) A review of BFN OI's indicated that several instructions do not have signoffs beyond the preparatory phase of accomplishing the procedure. ANSI N18.7-1976, section 5.3.4.1(2), recommends that checkoff lists be used for the purpose of confirming completion of major steps in the body of the procedures. Inasmuch as this is a recommended action, NSRS believes the minimum requirements are satisfied in this area.

b. Equipment Control and Status Programs

The review of equipment control and status systems included temporary alteration control, clearance control, shift turnover control, and equipment and valve status programs. Following is a discussion of each area:

(1) Temporary Alterations Control

The temporary alteration control program was reviewed to verify that the placement of temporary alterations were made with the proper level of evauation and authorization and that the location of the alteration and reasons for the alterations were properly documented. In addition, implementation of the program was reviewed to determine that safety functions were not being bypassed.

- Too many TACF's
- USQD's
- Designate SR/Non-SR

Primarily, NSRS was concerned with the large number of temporary jumpers in place on all BFN units, the fact that the program does not ensure management that the provisions of 10CFR50.59(b) will be met, and the fact that the program does not ensure that adequate review will be provided for temporary alterations for non-CSSC systems that may have safety functions described in the FSAR. A discussion of each of these is contained in the following paragraphs:

(a) NUC PR's program to control temporary modifications to the plant was delineated in BF SP 8.2, "Temporary Alterations," and DPM 73011, "Control of Temporary Alterations." These administrative instructions allowed temporary alterations of critical structures, systems, or components (CSSC) without development of a written safety evaluation to prove that an unreviewed safety question (USQ) did not exist. 10CFR50.59(b) required a written safety evaluation be developed to justify that a USQ did not exist for changes to the facility from that described in the FSAR. Therefore, the program did not assure that the provisions of 10CFR50.59 would be met. The Plant Operating Review Committee was responsible for reviewing temporary alterations to CSSC systems prior to the placing of the alteration, but they did not develop the written safety evaluation.

A review of the oustanding temporary alteration control forms in the Shift Engineer's office indicated that two alterations had been made to a CSSC system which clearly appeared to be beyond the intent of the present program as follows:

TACF No.	Date	System
1-79-063	12/7/79	Control Bay Chiller IB - A hand valve and an air relief valve were added to the oil cooler.
1-79-064	12/7/79	Control Bay Chiller IB - A liquid indicator was replaced by a check valve in the vent and oil recovery system.

This had been recognized by NUC PR and DCR 1840 dated April 30, 1980, had been generated to formalize the alteration. This DCR was approved by EN DES on June 11, 1980, and an ECN number was assigned; however, as of the time of the review the ECN, which would include an USQ determination, had not been completed. This did not appear to be a timely resolution of an identified deficiency.

A review of the temporary alteration control form in the Shift Engineer's office revealed that there was a total of 251 temporary alterations in place on all three units on CSSC systems. Two hundred and thirty of these had been in place for periods longer than six months, and some of them dated back to 1976. Technically speaking, a written safety evaluation should have been prepared for each one of these justifying that a USQ did not exist, although they had been reviewed by PORC and were not believed to involve bypassing of safety functions. DPM 73011 dated November 5, 1980, and BF SP 8.2 dated February 18, 1981, required that for temporary alterations on CSSC systems which had been in effect for periods greater than 60 days, a DCR be submitted prior to the end of the 60 day period to make the temporary alteration

permanent. It was evident that plant management was concentrating their attention in this area. The Plant Manager had implemented an information program to keep him apprised of the oustanding alterations; and, apparently, the total numbers of temporary alterations in place have been reduced during the past several months.

(b) NSRS did identify alterations to systems classified as non-CSSC systems that were described in the FSAR and may have safety significance as follows:

TACF No.	Date	Description of Function Bypassed	Safety Function Described In FSAR
3-79-164	12/7/79	Automatic isolation of steam supply to the steam jet air ejector on low pressure signal.	FSAR Table 9.5.4 - The safety analysis takes credit for the function to prevent detonable concentrations from existing in the off gas system that when detonated could result in an increased offsite release.
Not Recorded	4/25/80	High temperature alarm on one control rod drive 34-17 on TR-87-7B.	FSAR 3.4.6.4 - The high temperature alarm is a key indicator for the detection of leaks in CRD piping.

While the present program does not require PORC review of these alterations prior to placement, it requires periodic review (at least quarterly); it appears to NSRS that these would be worthy of review. At the outset it appeared not to be too great a risk to allow the bypassing of one or two CRD high temperature alarms, provided increased surveillance of the recorders was initiated. NSRS feels that placement of the alteration which apparently bypasses the steam supply isolation automatic function to the steam jet air ejector system (1-79-063) would not

be prudent unless special provisions or instructions were provided to assure that the steam supply would be manually isolated in the event of low pressure.

Prior to leaving the site on February 6 and prior to leaving the central office on February 10, 1981, NUC PR personnel were asked to review these alterations and to discuss the validity of these findings.

(c) A detailed review of two additional alterations indicated that sufficient information was not available on the temporary alteration control form to determine the exact location of the temporary alteration. These alterations are listed below:

TACF No.	Date Issued	Description
1-78-050	61180	Raw water clean-up system (RWCU) disabled action of 49X relay.
2-77-024	9777	RWCU Pump 2B low pressure switch bypassed.

Terminal locations of the alterations were not specified on the TACF's; therefore, it was not possible to tell whether No. 1, No. 2, or No. 3 (i.e., entire function) of the pump motor thermotectors were bypassed on TACF 1-78-050. On TACF No. 2-77-024, the RWCU Pump 2B did not have a low pressure switch according to the drawings available to NSRS. However, neither of the functions were described in the FSAR.

(2) Clearance Procedures

BF SP 14.25, "Clearance Procedure," delineated the requirements for taking plant systems out of service for maintenance under the cognizance of the Shift Engineer to ensure safety of personnel working on or around such equipment.

NSRS concluded that the program for equipment clearances was satisfactory, however, it was not

evident that requirements were established to provide independent verification of placing and removing of tags as recommended in ANSI N18.7-1876.

c. Shift Turnover and Equipment Status Systems

NSRS reviewed the programs and their implementation for maintaining status of ECCS systems and components described in SP BF 12.17, and the shift turnover program described in SP BF 12.7 and concluded that these programs were adequate and were being effectively implemented.

d. Evaluation of Reactor Transient and Trips

DPM N73015, May 1980, and BF 12.8, "Unit Trip and Reactor Transient Report," prescribed a program for evaluating all unit shutdowns or any significant reactor transient. The analysis consisted of evaluating system and reactor transient performance and comparing them against baseline performance parameters and reporting to CO management all deviations, causes, recommendations to prevent recurrence, personnel involved, and whether personnel error was involved. It included review and approval at the plant level.

NSRS reviewed this program and its implementation and concluded the program was adequate.

e. Quality Assurance and Quality Control of Operational Activities

Technical Specification 6.2.A.8.a requires the NSRB to conduct an audit to verify conformance of unit operation to provisions contained within the Technical Specifications and applicable licensed conditions at least once each 12 months. This requirement was established in February 1980 and POWER QA&A Staff was given the responsibility for performing the audits. NSRS reviewed 1979 and 1980 audits and found that the following audits were conducted during that period:

Audit Report No.	Date of Audit	Scope
BF 80-03	6/23-26/80	Technical Specifications 4.6.G 3.6.H and 4.6.H Verified PORC review of QA program employee training program

Audit Report No.	Date	Scope
BF-80-SP-05	9/23-25/80	Audited Technical Specification Manuals in various groups
BF 79-04	6/48/79	TS 4.3.a, 4.3.C.2, 3.10.A.3, 3.10.A.4, 4.10.C.1, 4.10.C.2, 4.10.D.a, 4.10.D.c, 4.4, and 4.4.c

Although the required audits were conducted, it appeared to NSRS that a sufficient sample of activities were not reviewed to give POWER management reasonable assurance that the technical specifications and applicable license conditions were adhered to in 1980. Discussions with OPQA&AS personnel indicated that they accomplished what they could with the personnel resources available. They also stated that a requirements matrix had not been established to enable a systematic approach toward accomplishing audit of all requirements over a period of years.

The plant had implemented a program of surveys of operational activities as prescribed in BF SP 3.9, "Quality Assurance Compliance Determination." NSRS reviewed this program and the surveys conducted by the onsite QA groups during 1980. Surveys involving plant operations primarily involved verifying CSSC valve alignments and review of implementing programs for documenting operating activities.

Generally, quality control activities were conducted by the operating groups. These activities consisted of supervisor review of documents and records and independent verificatin of certain operational activities, such as temporary alterations placement and removal and critical valve alignments by Auxiliary Unit Operators or Unit Operators.

f. Operator Responsibility

The licensed Senior Operator and licensed Operator responsibilities and authority were delineated in a number of DPM's and BFN SP's. The documents were reviewed to verify conformance with ANSI N18.7-1976 IEIN No. 80-06, and IE supplement No. 1 to IEIN 80-06. In addition, discussions were held with several licensed Senior Operators and licensed Operators to verify that they were aware of their responsibilities and authority.

NSRS concluded that the documents clearly defined the responsibilities Operators and Senior Operators and that personnel interviewed were aware of them.

g. Plant Tour

The reviewer conducted a tour of portions of the auxiliary building and of the three control rooms and found housekeeping in the areas to be satisfactory.

h. Documentation of Operational Activities

The reviewer examined selected operations journals, clearance logs, temporary alteration control forms, shift turnover records, scram reports, licensee event determination forms, surveillance data sheets, and system status files to verify that programs had been implemented which will assure that operating evolutions and events will be documented.

5. Communications

Mechanisms had been established to ensure the flow of verbal information between the CO and the plant; and between the plant manager and his key personnel as follows:

a. The Assistant Director (Operations) conducted a daily telephone conference call with the Plant Managers with his CO staff present.

b. The Assistant Director (Operations) periodically conducted a general tour of each plantsite or traveled to a site to resolve special problems.

c. The Plant Manager conducted general plant tours and held periodic meetings, some of which are described below:

(1) Daily coordination meetings, with Plant Supervisors, to discuss the activities and events of the past 24 hours and activities planned for the next few days.

(2) Periodic meetings with Plant Supervisors to discuss:

(a) Existing alarms in the control rooms

(b) Security degradations

(c) Radiological incidents

(d) Housekeeping

(e) Status of CAR correction

(f) Status of implementation of DPM's and N-OQAM changes

Mechanisms had been established both at the CO and the plant to ensure that correspondence was received by personnel with a need to know and that responses were generated on a timely basis.

Mechanisms had been established to keep the Operations personnel apprised of changes to the facility, changes to procedures, changes to Technical Specifications, and operating experience on a timely basis.

6. Surveillance Testing

The requirements for a surveillance testing program to assure that systems were operable as required by the Technical Specification were contained in BF SP 7.4, "Activity Control - Surveillance Testing," BF SP 17.9, "Surveillance Requirements Program," and BF SP 21.4, "Schedule System - Technical Specifications."
The program was reviewed to determine the adequacy of the:

a. Scheduling system

b. Verifications that testing is completed on a timely basis

c. Reviews by cognizant reviewers

d. Mechanisms for resolution of discrepancies

e. Effectiveness of the implementation of the program

The program and its implementation appeared satisfactory.

H. MAINTENANCE

The Director of NUC PR had been assigned responsibility for QA in plant maintenance and had assigned the Plant Manager responsibility for the maintenance program at his plant. Requirements for the safety-related portion of the nuclear maintenance program had been established in the N-OQAM, DPM, and the Plant Manager's SP's. The maintenance program provided for both preventive and corrective maintenance activities. Virtually all maintenance was

performed onsite by NUC PR personnel. The Plant Maintenance Staff performed maintenance on operating units and participated in major outage maintenance activities as required. The Outage Staff performed or directed maintenance on units shutdown for refueling or major maintenance. Power Systems Operations personnel provided maintenance and calibration services for both measuring and testing equipment and also for installed equipment as assigned. Assistance could be obtained from the POWER service shops and from outside contractors as necessary. All organizations that provided support were required to comply with applicable NUC PR requirements, as implemented by the SP's and plant instructions.

The nuclear maintenance program must satisfy the requirements of the Technical Specifications; of ANSI N18.7-1976, to which TVA was committed through Regulatory Guide 1.33 in the Topical Report; and of six ANS standards incorporated by reference in ANSI N18.7-1976. Applicable portions of the program must comply with 10CFR50, Appendix B, and the TVA QA Topical Report (TVA-TR-75-1A). Additional commitments had been made to the NRC in response to bulletins, orders, and other regulatory correspondence.

The nuclear maintenance program for BFN appeared generally adequate to meet regulatory requirements. Certain weaknesses and strengths are discussed below.

1. The policies, directives, organizational documents, and administrative procedures that establish the nuclear maintenance program were reviewed against regulatory requirements and compared to each other for consistency. Some of these documents were being revised or upgraded to reflect latest requirements and management decisions.

2. Program documents were reviewed and found adequate to meet regulatory commitments with exceptions as follow:

   a. The N-OQAM, Part II, Section 2.1, "Plant Maintenance and Repair," was found deficient with respect to certain committed requirements of ANSI N18.7-1976 and subordinate ANS standards. The CO QA Staff was processing for approval a revision of this section to upgrade or add requirements for the following:

      (1) Installation and protection of replacement equipment

      (2) Review and approval of vendor manuals and other information incorporated by reference in plant instructions

      (3) Evaluation of equipment failure

b. DPM's N74M7A and BF76M12 provided management policy with respect to the site Outage Staff organization. These DPM's, which were incorporated by reference into the plant SP's as the Outage administrative controls, were somewhat out of date. BF76M12 prescribed a QA staff function that no longer existed in the Outage organization. The Outage organization charts in both DPM's were out of date.

c. With respect to inspections by an independent inspection agency, requirements of section XI of the ASME Boiler and Pressure Vessel Code were not being completely met for inspection of modifications, repairs, and the results of inservice tests and examinations of ASME class 1, 2, and 3 components at BFN. Heretofore, TVA had allowed a CO section to act as the independent inspector for inservice activities covered by the ASME Code. However, in December 1979 TVA had received a letter dated December 12, 1979, from L. S. Rubenstein to H. G. Parris, "Inservice Inspection Program at Sequoyah" (L02 79121833). This letter stated that inspections by an ANI would be required for all TVA nuclear plants. A program to provide such services for BFN was in preparation at the time of this review, but the planned target date of March 1, 1981, for implementation would not be met. Because of the heavy impact of extended refueling outages scheduled for units 1 and 3 in 1981 and the considerable amount of effort required to complete alteration of procedures and indoctrination of personnel, implementation of the Section XI program could be delayed to some extent.

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During review of maintenance implementation (discussed below), it was noted that requirements for control and tracking of trouble reports (TR's), as defined in the N-OQAM and SP's, were stated in minimal language. A forthcoming revision to the N-OQAM, part II, Section 2.1 (discussed above), contained upgraded requirements establishing controls for such activities as the review of hold points, the definition of maintenance emergencies, and provision for special controls for such occasions. This seemed to be consistent with the significance of TR's in maintenance.

The general maintenance philosophy in NUC PR, as expressed in plant SP's and as supported by the NUC PR organization along functional lines not only satisfied regulatory requirements but appeared to provide the basis for a program of excellence. Functionally, the site staff was responsible for the performance of maintenance activities. As required, the Outage Management Branch provided manpower to assist in the implementation of maintenance. The

Nuclear Maintenance Branch and other CO branches developed program requirements and provided technical support to the plant as needed. The Nuclear Maintenance Branch and other CO branches had been assigned specific areas of responsibility that supported the major objectives of the maintenance program.

It was observed that strong emphasis had been placed on improving maintenance in the recent upgrading of management and supervisory organizations at the site, such as:

(1) Assigning responsibility for maintenance to an Assistant Plant Manager

(2) Creating assistant section supervisors and (temporary) general foremen to increase administrative capabilities and improve supervisor-to-craftsman ratios

3. Implementation of the nuclear maintenance program was verified in part by discussions with NUC PR personnel and by review of selected maintenance records. The program was judged to be adequately managed with some weaknesses.

A review of selected CSSC maintenance records including 20 trouble report/maintenance data packages--five each from the three plant maintenance sections and five from Outage--was made. The review indicated that maintenance activities had been conducted and documented properly with the following exceptions.

a. Conduct of Maintenance

Three trouble reports referenced the plant policy (BF SP 6.1) instead of an approved maintenance instruction as the authorized work instruction.

TR#	Date	Activity Performed
151066	2/12/80	Adjusted setpoint on 1-PRV-63-513
166681	10/6/80	Replaced roller and gasket on FCV-63-11A
189642	11/19/80	Repaired C1 RHRSW pump air release valve

b. Traceability of Materials

A power supply amplifier for a CSSC instrument was replaced with a unit taken from the instrument calibration lab. Repair records did not demonstrate traceability.

TR#	Date	Affected Instrument
189325	12/11/80	2-FT-74-56

c. Document Review

Five TR's showed no indication of a QA review of completed data.

TR#	Date	Subject
137205	4/12/80	Rebuilt RHRSW pump A1
103659	8/12/80	Replaced HPCI rupture disk
166681	10/6/80	---
104760	11/11/80	----
189642	11/19/80	Repaired float valve for C1 RHRSW pump

Item a above, failure to perform safety-related maintenance in accordance with approved procedures, was in noncompliance with the administrative requirements of the BFN Technical Specifications. Judged against the completeness and adequacy of data documented by other maintenance sections, the items listed in a and c above indicated a localized, not a programmatic, weakness. A plant management representative stated that failure to obtain a QA review of this maintenance data had been previously recognized and was being corrected. A plant management representative stated that the use of a "shop" component (item b) to replace CSSC components was recognized as a quality-affecting deficiency if traceability had not been maintained. He said in addition that plant management was considering measures to ensure traceability.

Welder certification files for both mechanical maintenance and outage personnel were reviewed. There were no discrepancies noted. During review of completed maintenance data packages discussed above, NSRS verified that required inspection hold points were signed off and that CSSC weld data appeared complete and adequate.

The reviewer pointed out minor deficiencies in control of the welding DPM and in locating certain weld data sheets to plant management.

Equipment history files were reviewed to verify entry of selected maintenance actions in regard to CSSC components. The histories appeared to be up to date in two of three cases. However, usefulness of the data for general trending reviews was questionable because of incompleteness or format. However, plant management stated that plant management and supervisory personnel were being apprised of repetitive failures of reportable equipment through the NRC's LER system. In addition, the maintenance supervisors and many of the crafts personnel had had extensive experience with BFN maintenance activities, so that many recurrent problems could be recognized without resort to a maintenance history. NUC PR personnel had been making preparations for assimilating maintenance data in an automated data base from which more valid trend evaluations could be made.

Onsite levels of manpower and experience appeared to be adequate for necessary maintenance activities, except that both Outage and Plant Maintenance Sections had experienced great difficulty in attracting and retaining engineers in the less experienced (SD-3) positions. It was verified that CO management had been very concerned about this weakness--several engineering positions had been recently upgraded to SD-4 in each Maintenance Section in the plant staff to improve recruiting and retention of qualified personnel.

Management controls and implementation records of the scheduled (preventive) maintenance program were reviewed. The accessibility and usefulness of the system, which provided semiweekly schedules and printed TR's for work control, appeared to be excellent. Recent weekly reports to the Plant Manager indicated that the schedule had been met consistently.

4. Program awareness of NUC PR personnel was examined during discussions with a broad cross section of NUC PR personnel, both onsite and in the CO. It was concluded that these personnel had generally a very clear understanding of the responsibilities of their position and also an awareness of the resources and capabilities of their own organization and those with which they interface.

I. PLANT MODIFICATIONS

The evaluation of the plant modifications program included an assessment of the program establishment, adequacy, and implementation

to ensure that the as-built quality of the plant was not degraded.

The evaluation also included a review of the plant modifications program relative to the establishment of priorities, scheduling of work, control for the removal of equipment from service, and control for returning equipment to service following the implementation of modifications. The review of the later selected areas of the program indicated that programs have been established and are implemented to assure an adequate level of management control and supervision.

The review of work plan implementation and establishment of plant modifications control identified program weaknesses and some serious deficiencies which, if not corrected, could lead to problems with safety and regulatory agencies. The details section includes examples of plant modifications to illustrate the program problems identified. The examples were found by reviewing approximately 55 ECN's and DCR's.

1. N-OQAM contains a list of critical systems, structures, and componenets (CSSC) with the selection of equipment being based on the following definition:

   > Those items that are necessary to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe condition, and the capability to prevent or mitigate the consequences of an accident which could result in potential offsite exposures to individuals in excess of a significant fraction of those specified in 10CFR100.

   The means by which certain activities were handled utilized as the first step a determination if the affected system, structure, or component was contained in the CSSC list. Some of these activities included:

   a. Temporary alterations

   b. Plant modifications

   c. Field change requests

   d. Setpoint changes

   The CSSC list was used to determine the level of management review and to form the basis for applying the provisions of the QA program during the performance of key activities. The N-OQAM section on plant modifications (part II, section 3.2,

paragraph 5.5) assigned responsibility by stating, "The plant superintendent shall be responsible for ascertaining that the CSSC lists (Appendix A of this manual) are revised, if necessary."

Some examples of safety-related modifications implemented into the plant but not added to the CSSC list include:

a. Recirculation pump trip breakers

b. Recirculation pump trip logic

c. Cooling tower vacuum breaking system

d. Cooling tower lift pump trip system

e. High pressure coolant injection system outboard isolation valve bypass valve

NUC PR has taken steps to address the above situation by issuing DPM N79A5 entitled "Critical Structure, Systems, and Components (CSSC) Review Committee" with the latest revision dated April 18, 1980. This DPM designated the committee members and responsibilities which included the requirement that submittal of any requests for addition or revision to the CSSC list in the N-OQAM shall be referred to the committee for consideration.

Subsequent action has included the appointment of a subcommittee to review the methods to keep the CSSC list updated to reflect plant modifications.

The committee has also discussed whether the BFN CSSC list is complete in the present form and specifically if items such as instrumentation and controls and safety-related doors should be added to the CSSC list.

Although the above actions are positive steps to assure that the CSSC list is up to date and maintained in that condition, the lack of timely correction of the problem indicates less than an adequate level of attention by the appropriate individuals.

2. The TVA QA Topical Report (section 17.2.3) described the modification control process. Specifically, section 17.2.3.3 states, "Proposed modifications to the safety-related CSSC (except nuclear fuel and fuel-related components) shall be forwarded to EN DES for review and approval." The POWER QA Plan also described this control in the procedure entitled "Modification Control" (OP-QAP-3.1) and stated in section 5.1.3 under "Responsibilities of P PROD," "Submit to EN DES for a design review and safety evaluation proposed safety-related

modifications (other than to fuel assemblies and core components) which change the configuration or function of CSSC items. Modifications which do not affect the plants' configuration or function may be handled by P PROD rather than EN DES."

The BFN N-OQAM, part II, section 3.2, paragraph 2.0, "Review and Approval of Change Requests," described the Plant Superintendent's determination of whether the proposed modification was:

a. Non-CSSC related

b. CSSC related and

(1) Nonsafety related or

(2) Safety related

The N-OQAM delineated the means by which this determination was made. A random sample of DCR's was reviewed to assure that this determination was being made correctly with the following results:

a. Modifications were made to the unit 1 refueling platform (CSSC list item 17.2) and completed in December 1979. The modifications were accomplished by DCR 1613 and involved the installation of thermostatic controlled vent fans and air filters. The modification was approved onsite without EN DES review and approval.

b. Modifications were made to the standby liquid control system in all three units in 1979. The modification changed the CSSC portion of the standby liquid control system (CSSC list item 14.4.6). The modifications were accomplished by DCR 1714 and were not reviewed and approved by EN DES.

The above examples apparently resulted from improper determination of CSSC equipment and/or failure to follow written procedures.

3. 10CFR50 delineates that the licensee review requirements for proposed changes, tests, and experiments. Specifically, 10CFR50.59(a)(1) states: "The holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change,

test, or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question."

Relative to the specific requirement to make a written safety evaluation to determine whether an unreviewed safety question exists, for making changes to systems described in the SAR, the plant modifications program procedures failed to specifically require EN DES review and approval of changes to systems described in the SAR.

The method for determining whether the proposed modification is safety related and subsequently sent to EN DES for review and approval is addressed in the N-OQAM, part II, section 3.2, paragraph 2.1.2 by stating: "THE APPROVAL OF THE PLANT SUPERINTENDENT IS REQUIRED FOR ALL PROPOSED CHANGE REQUESTS TO PLANT SYSTEMS AND EQUIPMENT ASSOCIATED WITH THE CSSC. The plant superintendent shall determine whether the proposed modification is safety-related. The plant superintendent shall automatically declare that a modification is safety-related[5] when the modification is to be made to a critical structure, system, or component (CSSC) and the modification involves (1) a change to a TVA drawing; (2) a change to a vendor's manual; (3) a change to a vendor's drawing; (4) a change to a material specification of a CSSC component; (5) the opinion of the plant superintendent that plant safety might be adversely affected by the modification; or (6) CSSC replacement parts and components that cannot be purchased by part number, by industry standards, or by the original purchase specification or where NUC PR cannot ascertain (by vendor certification or other means) that the replacement part or component is functionally equal to the original part or component."

Footnote:
[5]All technical specification requirements for a safety review of modifications to FSAR systems are considered to have been satisfied when the plant superintendent declares a modification is safety-related in accordance with these criteria and the proposed modification is reviewed and approved by EN DES."

BF SP 8.3 also addresses the method by stating: "The Plant Superintendent shall determine whether the proposed modification is safety-related and shall automatically declare that a modification is safety related when the modification is to be made to a CSSC and the modification involves:

a. A change to a TVA drawing

b. A change to a vendor's manual

c. A change to a vendor's drawing

d. A change to a material specification of a CSSC component

°The opinion of the Plant Superintendent that plant safety might be adversely affected by the modification

°CSSC replacement parts and components that cannot be purchased by part number, industry standards or by the original purchase specifications or where it cannot be ascertained that the replacement part or component is functionally equal to the original part or component."

Two points must be made regarding the above procedures:

a. Both procedures specify that the Plant Superintendent shall declare that a proposed modification is safety-related when the modification is to be made to a CSSC and the modification involves one or more other specific requirements. This procedural requirement also indicates that a proposed modification made to a CSSC that does not involve any of the other criteria is not automatically declared to be safety related.

b. The N-OQAM procedure stated that the safety reviews of modifications to FSAR systems are considered to have been satisfied when the Plant Superintendent declares a modification is safety related in accordance with the N-OQAM criteria and the proposed modification is reviewed and approved by EN DES.

The existing written procedures did not include a mechanism for the safety review of modifications to FSAR systems if the proposed change is not declared safety related.

The review of implemented modifications confirmed that the procedures failed to assure that a written safety evaluation is made to determine whether an USQ exists for changes to systems described in the SAR.

Examples identified include:

(1) DCR 2160 relocated main steam line sensing lines from line A to line D. This change resulted in changes to SAR drawings. The work plan did not include an USQ determination.

(2) DCR 1007 modified a drain line on the RCIC system. This modification resulted in a revision to an

FSAR drawing. The work plan did not include a written USQ determination.

4. On August 1, 1980, BF SP 8.3, "Plant Modifications and Work Plans," was revised. This revision eliminated the process by which the Plant Superintendent could allow the proposed modification to be reviewed and approved onsite. In accordance with the August 1 revision, all proposed modifications must be sent to the CO for processing. The BF SP 8.3 is the only procedure with this requirement. Individuals interviewed at the CO stated that all DCR's are sent to EN DES for review and approval prior to implementation.

   A number of DCR's were processed per the requirements of the "old" procedure and have not been physically implemented into the plant. Corporate or site management should define the means by which these DCR's will be handled; specifically, will they, for example, be sent to the corporate office for submittal to EN DES for review and approval?

5. The Outage Group at BFN used a system of placing partially worked work plans not being actively implemented at the present time in a "hold" status. Approximately 75-100 work plans were in "hold." These work plans were in various stages of completion with some being physically complete in the plant.

   Partially worked work plans were placed in "hold" without the as-constructed drawings or plant procedures being revised to reflect physical configuration of the plant. Realizing that in the majority of cases the partially implemented system was not "functional," Operations personnel should maintain an awareness of the configuration in "hold" because of the potential affect on plant safety and operational reliability.

   Based on conversations with Outage personnel, the current policy is that the Outage Director must approve the placement of a work plan in "hold." In a memorandum dated February 5, 1981, the Outage Management Branch Chief had directed the Modifications and Outage Support Supervisor to review outstanding ECN's and DCR's for BFN. Specifically, item 4 of the memorandum stated, "Identify ECN's for which partial field work has been performed."

   The above actions are good initial steps for managment to ensure that: (1) a minimum number of work plans are placed in "hold," (2) a minimum number of work plans are maintained in a "hold" status, and (3) review and evaluation of work plans in "hold" is accomplished to identify and document any potential safety/operational complications.

6. BF SP 8.3, "Plant Modifications and Work Plans," presents the methods and requirements for making modifications to the physical facilities of BFN. In the instruction portion of the SP (page 17), the following requirement is stated:

> "All modifications to the on-site electrical distribution system shall include an analysis of the added load to ensure the capability of the off-site and on-site electrical systems to maintain system voltages are not reduced."

The SP does not specify the individual or group responsible for performing this electrical analysis.

The SP does define responsibilities for the QA Supervisor as follows:

°Performs preliminary and final technical and procedural review of modification requests and implementing documents to ensure that all quality requirements are complied with in accordance with this and other applicable plant instructions.

°Provides support to the Quality Control Inspector for all modification work.

The supervisor responsibilities could easily be assumed to include a review to assure that the electrical analysis is performed and documented.

SP forms utilized in the work plan to assure that the accomplishment of tasks were documented by signatures did not specify the requirement for an electrical analysis.

Work plans for the low pressure coolant injection (LPCI) system and emergency feeder for the LPCI motor-generator (MG) sets were reviewed in an attempt to locate an analysis. None were included in the electrical work plans reviewed. The requirement for the electrical analysis could be construed to imply that EN DES will perform the analysis. The use of the SP to delineate an EN DES requirement is not correct. The requirement could also be construed to mean that the EN DES USQ determination will satisfy the electrical analysis requirement. If this is the case, the need for the requirement in the SP is questionable. Another important point was the requirement that plant personnel perform an electrical analysis for DCR's approved onsite (EN DES review and approval was not required). Conversations with individuals during the review failed to identify anyone who was responsible for performance of this analysis.