Safety Evaluation Report Restart of K-Reactor Savannah River Site

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1. INTRODUCTION

In 1981, in the aftermath of the events at Three Mile Island, the DOE described certain deficiencies at SRS in a very thorough, 15-volume study of DOE reactor facilities. In 1987, a group of reactor safety experts assembled by the National Academy of Science (NAS) reviewed DOE reactor operations and found that little had been done to correct the deficiencies noted in the 1981 report. Further, the NAS review indicated that reactor power levels had been reduced to half power due to technical issues that had recently surfaced. Contractor management had not been aggressive in dealing with safety questions raised.

In August 1988, with both the L- and K-Reactors in an extended outage, an incident occurred at the SRS P-Reactor that raised questions about the operating contractor staff's ability to understand and properly respond to abnormal events. Pending a review of the incident, the P-Reactor was also shut down.

In November 1988, the DOE issued the "Savannah River Plant K-Reactor Restart Strategy" (also known as the "Charlotte Criteria"). This document was the result of a diagnostic evaluation of issues raised, not only from the August 1988 event, but also from earlier safety reviews. It contained requirements to be met before restart.

On April 1, 1989, the operating contract for SRS was transferred from E.I. Du Pont de Nemours & Company (DuPont) to the Westinghouse Savannah River Company (WSRC). In preparation for this transition, WSRC performed a corporate safety assessment of the reactor design and operations. WSRC published the results of their assessment in the "Westinghouse Independent Safety Review of Savannah River Production Reactors" (WISR), April 1, 1989. This document also included a list of recommendations for improvements to the reactor operations, design and management for both before and after restart.

On May 19, 1989, the Secretary of Energy directed a number of specific management and organizational changes within DOE. These changes, which are discussed in Secretary of Energy Notice SEN-6-89, were specifically designed to establish line organizations that are fully responsible for their programs. Among the changes discussed in SEN-6-89, was the establishment of the Savannah River Restart Office (SRRO) at DOE headquarters in Germantown, MD and the Savannah River Restart Special Projects Office (SRSPPO) at the Savannah River site. These organizations were tasked with activities in support of the safe and timely restart of the Savannah River reactors, including the evaluation of programs developed to correct safety and management problems at SRS. The results of the SRRO/SRSPPO evaluation are included in this SER.

This SER focuses on those issues required to support the restart of the K-Reactor. Issues related specifically to the L-Reactor will be addressed in a supplement to this SER, which will be issued prior to restart of that reactor.

This SER provides the safety criteria for restart and documents the results of the staff reviews of the DOE and operating contractor activities to meet these criteria. To determine the issues of concern to be addressed by this SER, the

staff first reviewed deficiencies identified in reports, such as the NAS Report (November 1987), the Charlotte Criteria and the WISR to identify general areas of concern that needed to be addressed to support reactor restart. The DOE Management and Program Oversight issue, for example, generally summarizes the DOE onsite technical vigilance and monitoring (TV) issues discussed in the Charlotte Criteria. Specific management issues, such as safety culture, were also identified based on experience with commercial reactors (e.g., Peach Bottom and the Tennessee Valley Authority plants), where issues such as these have been shown to be the root causes of many long term deficiencies. Some issues were identified on the basis of a comparison between the draft DOE Safety Policy (which identifies those areas to be considered to ensure safe operation of a DOE facility) and previously identified areas of concern at SRS.

To develop the restart criteria for the issues discussed in this SER, the SRRO and SRSP0 staffs relied, when possible, on commercial industry codes and standards and on NRC requirements and guidelines for the commercial nuclear industry. However, because of the age and uniqueness of the Savannah River reactors, criteria for the commercial plants were not always applicable. In these cases, alternate criteria were developed. The restart criteria applicable to each of the issues are identified in the safety evaluations for each issue. The restart criteria identified in this report are intended to apply only to restart of the Savannah River reactors. The long range goal for the Savannah River reactors is to have them meet criteria that are equivalent to criteria used for older, commercial reactors.

Following the development of the acceptance criteria, the DOE staff and their support contractors evaluated the results of the DOE and operating contractor (WSRC) activities to meet these criteria. The results of those evaluations are documented in this report. Deviations or failures to meet the requirements are either justified in the report or carried as open or confirmatory items to be completed and evaluated in supplements to this report before restart.

2 MANAGEMENT ISSUES

2.1 DOE Management and Program Oversight

2.1.1 Background

As a result of the Department of Energy's (DOE's) failure to recognize the developing severity of the problems at the Savannah River Site (SRS), confidence in the DOE reactor-oversight process was lost. On May 19, 1989, the Secretary of Energy directed in Secretary of Energy Notice 6-89 (SEN-6-89) (Reference 1) that a number of management and organizational changes be made to (1) improve the DOE oversight process, (2) establish direct line responsibility, and (3) focus Departmental resources on the safe and timely restart of the SRS reactors. These changes involved, in part, the establishment of the Savannah River Special Project Office (SRSP0) at SRS and the Savannah River Restart Office (SRRO) at DOE headquarters. Both groups report directly to the Deputy Assistant Secretary for Nuclear Materials.

The Director of SRSP0 was assigned the responsibility of directing the contractor's efforts on matters directly related to the managerial, technical, operational (including training), and scheduling aspects of the safe restart of the K-, L-, and P-Reactors.

SRSP0 staff currently consists of about 55 management and technical personnel and 17 administrative personnel. The organization is divided into four groups: (1) the Technical Support Division (TSD), which is responsible for overall reactor restart program integration and support activities including planning, issue management, DOE training and procedures, the Reactor Safety Improvement Program, environmental activities, and emergency planning; (2) the Reactor Operations and Training Division (ROTD), which is responsible for oversight of the day-to-day activities at the reactors including operations, maintenance, modifications, and testing as well as the oversight of contractor procedures, training, and response to events; (3) the Safety Oversight Division (SOD), which is responsible for oversight of quality assurance, health physics, and the independent oversight of design, management, conduct of operations, reactive appraisals and event follow-up; and (4) the Reactor Engineering Division (RED), which is responsible for review of the design and analysis of the facility including modifications, system testing, and inspections.

SRRO's staff of approximately 17 technical and management personnel is divided into three divisions: (1) the Environmental, Occupational Safety, and Emergency Preparedness Division, which is responsible for oversight of environmental requirements and compliance, environmental impact studies/statements, worker/public health and safety, emergency preparedness, and quality assurance; (2) the Plant Readiness Division, which is responsible for issues related to reactor operations, training, standards and procedures, operational tests, maintenance and technical support; and (3) the Project Control Division, which is responsible for the coordination of the Safety Evaluation Report (SER) for reactor restart and the oversight of budget, schedules, milestones, and issues management.

The Director of SRSP0 has the line responsibility for the oversight of the SRS reactors. For this reason, the evaluation of DOE's oversight of the SRS reactors focuses primarily on the efforts of the SRSP0 organization. The review and evaluation of SRSP0, which was performed by the SRRO staff, was divided into five discipline areas: (1) Safety Culture, (2) Documentation and Formality, (3) Involvement in Reactor Operations, (4) Training and Technical Proficiency, and (5) Inspections and Audits. The restart criteria evolved largely from the November 1988 Savannah River Plant K-Reactor Restart Strategy (Reference 2), which is also known as the Charlotte Criteria, and are based on those developed for similar NRC oversight activities. The restart criteria also contain supplementary requirements based on corrective action plans from commercial plants designated by the NRC as "problem plants."

2.1.2 Restart Criteria

The criteria for determining the adequacy of DOE oversight activities for restart are described below.

SRSP0 shall oversee the efforts of its contractors at SRS to ensure safe reactor operation and satisfaction of production objectives without violation of applicable operational, environmental, safety, health, or security standards. This shall be accomplished by providing clear direction to the contractor and by verifying acceptable performance based on inspections of reactor, engineering, and facilities operations. SRSP0 oversight responsibilities for each of the five discipline areas include:

(1) Safety Culture

- (a) The DOE safety culture shall reflect the philosophy that protection of the environment, safety, and health is to be more heavily weighted than production. There should be formally stated policy guidance to this effect. Performance elements for all technical, professional, and management personnel at SRSP0 shall emphasize the importance of safety. In addition to stated policies and performance elements, SRSP0 management shall use additional means (e.g., performance awards, staff meetings, training) to identify and communicate the cultural values of (1) adherence to safety, environment, and health requirements and policies, (2) engineering excellence, and (3) individual accountability for performance.
- (b) The DOE safety culture shall be expressed to the contractor through clear and specific guidance, and commitment to this principle shall be further evidenced by a willingness on the part of DOE management and staff to challenge the contractor to demonstrate conformance to stated policies, laws, and regulations. In addition, the emphasis on safety, environment, and health shall be clearly reflected in the operating contract and in the process of awarding fees. DOE procedures shall also clearly state actions to be taken when unacceptable performance is discovered during daily oversight activities. DOE line accountability for performing these actions shall also be clearly stated.

(2) Documentation and Formality

- (a) Formally documented procedures and policies that assign clear lines of responsibility shall be developed for SRSP0. The following list includes the minimum issues for which formal procedures and policies should be developed and implemented:
 - o assessment and appraisal process
 - o tracking and resolution of identified problems and issues
 - o receipt and dissemination of notification of events
 - o receipt and dissemination of lessons learned on issues applicable to the SR reactors
 - o formal actions in response to event reporting requirements
 - o formal process for review of new or revised orders, procedures, and policies from inside and outside DOE
 - o formal safety evaluation process, including review of unresolved safety questions
 - o formal process for feedback to employees on performance and the resolution of employee concerns
- (b) SRSP0 procedures shall clearly define line responsibilities for identifying and pursuing problems at the SR reactors. In addition, the procedures shall clearly state actions to be taken if contractor response is inadequate for the situation.

(3) Involvement in Reactor Operations

SRSP0 management and staff shall be directly involved in the daily operation of the SR reactors. This involvement should be evidenced by the following:

- (a) SRSP0 facility representatives stationed at each of the reactors
- (b) a formal plan for unannounced SRSP0 management walkthroughs of the reactors, including a process for recording the time spent, the reactor inspected, the findings from the visit, and the actions being taken to effect contractor resolution of significant findings
- (c) a plan for enhancing the technical capability of the SRSP0 staff and management
- (d) defined SRSP0 management and staff roles and duties for coverage of contractor activities, including backshift and deep backshift coverage where appropriate
- (e) clearly defined and documented SRSP0 stop work authority
- (f) formal and timely SRSP0 review and approval process for unreviewed safety questions
- (g) oversight of contractor review groups

- (h) formal process for review of contractor compliance with DOE orders
 - (i) DOE presence during major activities at the reactors
 - (j) formal procedures specifying actions to be taken when unacceptable performance is discovered during daily oversight activities
- (4) Training and Technical Proficiency
- (a) SRSP0 shall have a plan for developing an integrated staff that is proficient in the technical areas related to SR reactors. A formal plan shall be developed defining the technical capabilities required for the SRSP0 staff and describing how these capabilities will be met.
 - (b) By reactor restart, SRSP0 shall have a formal training program in place for technical personnel. This program shall include the following elements:
 - o health physics/radiation protection*
 - o duties, responsibilities, and authorities of staff technical positions*
 - o inspection, audit, and reporting procedures (required for selected personnel before restart, others may receive it post restart)
 - o unreviewed safety issue process*
 - o simulator training for selected staff
 - o general employee training (including contractor requirements while in the reactor area)
 - o plant systems
 - o new or revised procedures, policies, or orders (as they are issued)
 - o lessons learned at SRS or other facilities

(Technical personnel shall complete training in those areas indicated by an asterisk * before reactor restart.)
 - (c) SRSP0 technical and professional management shall complete training in the following areas before reactor restart:
 - o Institute of Nuclear Power Operations (INPO) publications and recommended practices
 - o team building
 - o surveillances, audits, and reporting
 - o reactor walkthroughs
 - o health physics/radiation protection
 - o unreviewed safety issue process
 - o lessons learned at SRS or other facilities
 - (d) A formal method for the staff to demonstrate the effectiveness of the training through oral, written, or practical exams shall be established for selected courses. In addition, a training

schedule shall be established, and records shall be maintained indicating attendance and satisfactory completion of the training.

(5) Inspections and Audits

- (a) SRSP0 shall establish and implement a formal inspection and audit program to verify operation of the reactors in accordance with established environmental, health, safety, operational, and security laws and requirements. The program shall include: (1) a reporting and tracking process for findings; (2) a mechanism to communicate the problems to the contractor in a timely manner; (3) a requirement for prompt dispositioning and timely closure of the identified issues; and (4) clearly assigned responsibility and accountability for follow-up of the issues by SRSP0 (defined group/person). SRSP0 shall maintain sufficient qualified staff to implement the program effectively.
- (b) Training in inspection and audits shall be provided to the staff members who will perform them. The inspection criteria shall be clearly defined. The inspector shall be familiar with the inspection criteria and with the procedures and/or standards applicable to the review area. SRSP0 procedures shall clearly state the actions to be taken when unacceptable performance is discovered during daily oversight activities.
- (c) SRSP0 shall have formal procedures for response and follow-up to reactor incidents determined to have potential significant impact in the areas of safety, health, environment, or operation. Thresholds for determining the appropriate levels for incident response and follow-up shall be clearly documented.

2.1.3 Evaluation

Safety Culture

When the SRS reactors were shut down in August 1988, the prevailing culture at DOE held that environment, health, and safety were secondary to meeting production requirements to support the national defense programs. On June 27, 1989, the Secretary of Energy announced a ten-point plan (Reference 3) to strengthen the environmental protection and waste management activities at DOE's production, research, and testing facilities. That plan included resetting the priorities of DOE such that environment, safety, and health would be more heavily weighted than production. In addition, the initiatives included modifying the criteria for awarding contractor fees to reflect increased emphasis on environment, safety, and health.

In a letter to Westinghouse Savannah River Company (WSRC) dated August 11, 1989 (Reference 4), the Award Fee Determination Plan for SRS was revised to allot a total of 58.25 percent of the total award fee dollars to environment, safety, and health requirements. This change was made to provide clear direction to the contractor regarding the increased emphasis on environment,

health, and safety in accordance with the Secretary of Energy's June 16, 1989, ten-point initiative.

In a letter, dated March 15, 1990 (Reference 5), DOE transmitted the SER restart criteria to WSRC. The restart criteria provide clear and specific guidance to the contractor regarding the safety culture by stating minimum standards that must be met before restart of the reactors. These standards are based primarily on safety and health considerations.

Most of the SRRO staff has worked in some capacity with the NRC or with commercial or navy reactors within the last 3 years. As a result, they bring the safety-conscious culture of those environments to SRRO. SRSP0 is also heavily represented by those with commercial and navy experience. However, a significant portion of the non-supervisory staff at SRSP0 have spent a number of years working in a culture with inadequate emphasis on safety and health. Therefore, most of the actions to encourage a stronger safety culture were focused on SRSP0 staff personnel.

In an effort to nurture the newly emerging safety culture and to encourage others who were reluctant to raise safety issues, the office directors for SRSP0 and SRRO issued a memorandum to their staffs and support contractors on April 13, 1990 (Reference 6). In this memorandum, the office directors requested that individuals inform them of any outstanding safety or technical issues that had not been properly identified.

Performance elements and standards that incorporate the increased emphasis on health and safety are being developed for both SRSP0 and SRRO. A review of several Performance Appraisal Plans (PAPs), DOE F 3430.7, indicated that many contained a performance element that stated that the individual will carry out environment, safety, and health (ES&H) functions as assigned, overview contractor performance regarding ES&H activities, and demonstrate a clear sense of safety consciousness while carrying out assigned responsibilities. SRRO personnel have a similarly worded performance element in their PAPs. Both organizations are committed to strengthening the wording of this performance element to include the concept that ES&H considerations are to be more heavily weighted than production. Pending revision of the PAPs, the criterion to emphasize safety in performance elements remains an open item.

On January 30, 1990, SRSP0 issued Administrative Instruction (AI) 102, "SRSP0 Organization and Responsibilities" (Reference 7). This document clearly identifies the general responsibilities and authorities for the office director of SRSP0 organization and for the four division directors. Other administrative instructions describing division responsibilities were developed; however, some of these instructions contained jurisdictional conflicts. To correct this problem, SRSP0 is revising AI-102 as AI-2, "Roles and Responsibilities" (Reference 8). Draft AI-2 has been reviewed, and the overlap of responsibilities among the various divisions has been removed. An approved final version is expected to be published in 1991 prior to restart. SRSP0 has identified, and is committed to publishing, a total of 50 instructions, guidelines, and procedures prior to restart. Pending final publication, the criterion to provide formal policy guidance and procedures that emphasize safety philosophy remains an open item.

Documentation and Formality

SRSP0 has initiated a procedures manual to establish a system of administrative controls and provide guidance to SRSP0 staff and their support contractors. Administrative Instruction (AI) 100, "Manual Organization" (Reference 9), defines the SRSP0 organization and details the organization of the procedures manual.

The manual establishes both AIs and Administrative Guidelines (AGs). AIs are the written instructions of the Director for SRSP0. These instructions govern the activities of the Director's staff and SRSP0 support contractors. AGs provide guidance concerning standards of conduct and methods for performing duties. AGs are not mandatory; however, SRSP0 staff and their support contractors are expected to use them unless reasonable justification exists for not doing so.

The manual is divided into five sections. The first section contains AIs and AGs that apply to the entire organization. Each of the other four sections is dedicated to an SRSP0 division and contains the procedures and guidelines applicable to that specific division. The five SRSP0 sections are:

- Section 1 - SRSP0 Administrative Instructions and Guidelines
- Section 2 - Technical Support Division (TSD) Procedures and Guidelines
- Section 3 - Reactor Engineering Division (RED) Procedures and Guidelines
- Section 4 - Reactor Operations and Training Division (ROTD) Procedures and Guidelines
- Section 5 - Safety Oversight Division (SOD) Procedures and Guidelines

AI-102 assigns lines of responsibility for each of the SRSP0 divisions. In addition, a more specific list of responsibilities is listed in each division's charter, which is also included in the manual. As previously discussed, this AI is being revised and will be reissued as AI-2.

Manual instructions and guidelines that specifically address the restart criteria include: (1) OG-450, "Operations Inspections" (Reference 10), and OG-451, "Maintenance and Surveillance Inspections" (Reference 11), which contain the guidelines for performing inspections and surveillances; (2) AI-103, "Management Walkthroughs" (Reference 12), which delineates responsibilities of SRSP0 regarding management walkthroughs of the reactors; and (3) AI-104, "SRSP0 Notification to the Contractor of an Adverse Condition/Stop Work Order" (Reference 13).

Additional instructions and guidelines, which are expected to address the restart criteria more fully, are being prepared and include: (1) AI-105, "Event Notification and Investigation" (Reference 14); (2) AI-107, "Operational Readiness Review" (Reference 15); (3) AI-112, "Commitment Tracking and Document Control" (Reference 16); (4) AI-119, "Issue Management" (Reference 17); and (5) draft AI-124, "Unreviewed Safety Questions" (Reference 18). These instructions and guidelines have been reviewed by SRRO personnel to assess satisfaction of the restart criteria. Those reviewed to date have been satisfactory.

Some of the restart criteria yet to be fully addressed by published instructions or guidelines include:

- (1) a formal review process for review of new or revised orders, procedures, and policies both inside and outside SRSP0
- (2) a formal process to address and resolve employee concerns

Procedures and guidelines for the above criteria have been prepared and are in draft form. SRR0 will continue to review SRSP0's progress in documentation of formal procedures. The results of the continued progress in meeting the restart criteria through development of appropriate procedures will be reported in a supplement to the SER.

Involvement in Reactor Operations

SRSP0 has completed its evaluation regarding the number of facility representatives at each reactor. A minimum of two facility representatives, who are responsible for monitoring and daily oversight, are assigned to each of reactors (K, L, and P). The SRSP0 facility representatives are part of ROTD. The remainder of the ROTD personnel are located at the C-Reactor area, close to the K-, L-, and P-Reactor sites.

SRSP0 has been involved extensively in both the L-Reactor flow test and the ultrasonic tests for the K-, L-, and P-Reactors. Twenty-four hour coverage of these tests was provided during fuel movement when deemed necessary. In addition, SRSP0 has reviewed and approved WSRC's fuel load procedure, which contained hold points for critical evaluations. Also, prior to fuel movement, OG-457, "Preparation for Refueling" (Reference 19), and OG-458, "Reviewing Activities" (Reference 20), were developed and implemented to ensure that WSRC was monitoring fuel load activities on a 24-hour basis.

OP-402 and OP-405 document the facility representatives responsibilities during shift coverage, including monitoring control room operations, charge and discharge operations, shift manager command and control, technical specification compliance, system status control, shift turnover, and verifying equipment operability. The facility representatives maintain a log to record important information such as reportable events, major accomplishments, significant problems, and the completion of shift turnover checklists.

ROTD has drafted OP-409, "K-Reactor Restart Evolutions" (Reference 21), to augment the K-Reactor facility representative staff with three additional representatives during restart. The procedure will define the standards used by the directors of SRR0 and SRSP0 to determine when augmented coverage is to be initiated and terminated. It will also specify management and staff roles, duties, and assignments during the restart, including provision for 24-hour coverage by reactor facility representatives and on-site presence of management during key restart periods.

OP-401, "Duty Officer" (Reference 22), and draft OP-404, "Event Notification and Occurrence Reporting" (Reference 23), provide guidance for reactor facility representatives in performance of their Duty Officer responsibilities

and for the timely notification by the operating contractor when off-normal occurrences are identified. OP-401 and OP-404 address the designation of Duty Officers; dissemination of notification of off-normal occurrences to the appropriate DOE personnel; tiered notification and action requirements based on anomaly significance; and decision-making requirements for authorization for restart, continued operation, or other corrective action.

OG-450 and OG-451 provide formal guidance for the performance of inspections and surveillances by SRSP0 staff at the SR reactors. These guidelines were modeled after the NRC's Inspection Plan for Commercial Nuclear Facilities. OG-450 includes guidance for the periodic inspections that are to be performed on daily, monthly, and bimonthly schedules. OG-450 also provides guidance for the minimum inspection requirements for the SR reactor facilities. The attachments to OG-450 and OG-451 provide specific guidance for a broad range of inspection and surveillance activities, and with implementation enhance the safe operation of the reactors.

AI-103 established the program SRSP0 uses to increase SRSP0 management presence at the reactor facilities. The instruction requires SRSP0 management to dedicate a minimum of 5 hours each week to perform unannounced reactor facility walkthroughs. The instruction also requires that managers should cover all shifts on a rotating basis. A Management Walkthrough Report to be submitted to the Director of SRSP0 within 2 working days after each walkthrough should include as a minimum the date, time, duration, and location of the walkthroughs, the activities observed, the results of the surveillance, and any followup actions taken or required. The Director of SRSP0 assigns responsibility for these followup actions.

Approximately 50 walkthrough inspection reports, covering the period from December 1989 through June 1990, were reviewed by the SRRO evaluator. It was concluded that the content of these reports fully complied with all AI-103 requirements. Findings from the walkthroughs are entered into and tracked using the SRSP0-developed and -operated issue management data system.

AI-104 established responsibilities and authorities regarding notification to the contractor to stop work on the reactors. If SRSP0 personnel perceive that the corrective actions taken by a contractor after being notified of an adverse condition are insufficient or inadequate, this deficiency shall be brought to the attention of SRSP0 management. If the Director of SRSP0 then determines the corrective actions to be inadequate or insufficient he/she may require the contractor to stop work. The requirements for establishing and removing a stop work order are clearly delineated in this instruction. SRSP0 managers have not had cause to issue a stop work order since January 30, 1990, the date AI-104 was published.

Actions to be taken when less significant unacceptable performance is encountered are discussed in administrative instructions on specific topics. AI-121 "Safety Evaluation Report Assessment Program" (Reference 24), requires that the four SRSP0 divisions provide the contractor with monthly inspection reports which track the findings from inspections in each division and specify the current status of each finding. These monthly inspection reports, which include the information from the individual walkthrough inspections, are

prepared by each SRSPD division. The four division monthly inspection reports and a summary section are then published as a single monthly report. Several of these monthly reports were evaluated and judged to be in full compliance with AI-121 requirements.

On March 30, 1990, the Director, SRSPD, established the position of Chief Test Engineer (CTE) in the SRSPD. The CTE is authorized to approve Startup Test Activities (STA) for DOE, including Reactor Test Identification Reports, Startup Test Procedures, and Startup Test Results. The CTE was established to support increased involvement of the SRSPD in the review and approval process of the Startup Test Program (STP). This action demonstrates a recognition of the SRSPD of the critical nature of the STP program.

Based on the above, the criteria for SRSPD involvement in SR reactor operations is acceptably satisfied.

Training and Technical Proficiency

SRRO has established a training program for its personnel in the following subjects: (1) SRS System Overview, (2) SRS General Employee Training, (3) SRS Radworker Training, (4) Safety Evaluations (5) DOE Legal Environment, (6) Selected DOE Orders, and (7) for supervisors only, DOE Supervisor Training.

The Secretary of Energy in SEN-6B-90, issued August 2, 1990, reiterated his intentions, previously stated in SEN-21-90, to ensure that training and staff development of DOE program managers, supervisors, and executives is comprehensive in scope, properly coordinated, and receives the resources necessary to achieve early positive results. The Secretary directed the newly created Office of Scientific and Engineering Recruitment, Training, and Development (TR) to:

- o Establish and conduct a mandatory, fast-track, standardized training program in DOE conduct of operations and management responsibilities for all incumbent SES managers involved with either line management or oversight responsibilities of technical programs or facility operations.
- o Direct the establishment and conduct of a mandatory, fast-track, standardized training program in fundamentals of DOE oversight, including the conduct of operations, for DOE program managers and field staff.
- o Compile and review all DOE technical (scientific and engineering) and executive training programs for DOE incumbents, and provide a recommendation as to training needs for DOE incumbents. Pending the completion of such efforts, immediately defer or cancel all DOE-sponsored training (except Tiger Team training), both direct and indirect, for DOE technical staff, managers, and executives.
- o Establish controls to prevent the formation or proliferation of redundant technical or executive training organizations and initiatives within DOE.

The need for a formal, well-conducted DOE-wide training program directed by TR was fully endorsed by the entire DOE SRS organization, which established an Office of Training (OT) to serve as liaison. The OT mission is to develop performance-based training requirements for all SRS personnel to support their proficiency in the execution of assigned duties. In early 1991, a manual will be issued to all divisions and offices at SRS that describes the procedures necessary to implement the site-wide training program. The goal is to integrate OT and SRSP0 training programs by the end of 1991.

The reactor restart training program, as described in AI-106, "Training" (Reference 25), represents the current on-going effort for enhancing SRSP0 managerial and technical staff basic skills and abilities and for satisfying the restart criteria. As a long-term objective, it is anticipated that the TR Headquarters-directed training program, in concert with the OT training program, will ultimately provide the continuing basis for an integrated SRSP0 staff proficient in technical areas related to SRS reactors and will include future skills maintenance and developments.

AI-106 defines required technical capabilities for each class of positions and describes how these capabilities are being developed. The AI establishes:

- o a training coordinator staff position
- o restart training requirements
- o preparation and completion of Training Requirements Records (TRRs)
- o training requirement fulfillment by equivalency
- o training completion certificates
- o evaluation of classroom training
- o maintenance of DOE-2 files and records
- o training schedules
- o DOE Form 3410.4 requirements
- o instructor qualifications
- o preparation and maintenance of job task analyses (JTAs)
- o job responsibilities
- o documentation and retention requirements

Specific training requirements are established for each class of position within SRSP0. Supervisors must evaluate and determine each staff member's specific training requirements for a position in relation to its duties and responsibilities. The supervisor then ensures that the training necessary for performance of a duty is completed prior to a given date. If the training is not completed as scheduled, the employee is prohibited from performing that duty without the close supervision of a qualified individual.

All SRSP0 technical and managerial employees are required to complete the following basic training courses:

- o SRS general employee training
- o SRS radiation worker training (general and 100 area specific)
- o guidance for conducting safety evaluations
- o INPO Observation Course

In accordance with the restart criteria, the SRSP0 managers are one of two groups with significant mandated restart training, inscluding:

- o INPO management seminar
- o Reactor facility walkthrough instructional tour and follow-up exercise
- o Team building course
- o Lessons Learned in the Operation of Nuclear Facilities course
- o Selected Emergency Response Organization (ERO) courses based on an individual's duties and responsibilities.

The SRSP0 K-Reactor facility representatives also have significant mandated restart training, including:

- o inspection course
- o introduction to the SR reactors course
- o introduction to component handling course
- o integrated start-up and shutdown course
- o simulator training
- o reactor physics and thermodynamics course
- o self-study of the Safety Analysis Report; reactor system self-study modules and walkdowns; topical reading assignment booklets; and selected DOE Orders, portions of the Code of Federal Regulations, Nuclear Industry Codes and Standards, INPO Good Practices and Guidelines, sections and chapters of the NRC Inspection and Enforcement Manual, and DuPont Standard Operating Procedures.

In addition to basic and mandated restart training requirements, each Division Director is permitted to specify enhancements based upon unique job responsibilities, Job Task Analysis, position descriptions, reviews of DOE Orders, and/or functional needs. These enhancements comprise a majority of the required training.

The above training requirements are consistent with the restart criteria and if present progress continues, it is anticipated that all the SRSP0 managers and facility representatives will meet their mandated restart training commitments. As reflected in draft AI-106, Revision 1, SRSP0 is to designate a subset of fully qualified individuals from the two groups, who will have the primary responsibility of overseeing the K-Reactor restart. This subset would be known as the reactor restart task force.

OP-403, "Personnel Training" (Reference 26), was the original publication that defined the training requirements for the ROTD facility representatives.

Because of the critical nature of their job, each facility representative must complete an oral examination when all elements of his/her training requirements record (TRR) have been completed. Records are maintained by the SRSP0 Training Coordinator, who must receive written notification of successful completion before credit for the course is given.

OP-403 does not require the ROTD facility representatives to complete the restart training regimen given to the operating contractor shift crews. The restart training regimen for operating contractor shift crews is designed specifically for reactor operators and supervisors and was designed to supplement their previous training with training on the new safety culture and on reactor fundamentals. The ROTD facility representatives on the other hand require training to assure proper oversight of the operating contractor's staff. This requires different knowledge levels and skills than is required for reactor operators. With respect to safety culture the facility representatives are primarily from either the navy nuclear program, the NRC, or the commercial nuclear industry and have brought with them the safety-conscious cultures of those environments. The training of SRSPo facility representatives focuses on the basic technical proficiencies required to understand the design and operational aspects of the Savannah River production reactors, and the performance training needed to prepare them for the inspection, assessment and auditing tasks required by the positions. The first category includes classroom training on systems, integrated plant operations, core physics, and thermalhydraulics; simulator training; and self study on system design information supplemented by time in plant to provide familiarity with plant and system physical configuration. The latter category includes the general employee training, radiation worker training, and security training required to meet the contractor requirements for unrestricted facility access as well as instruction in inspection and assessment techniques. Oral examination is used to verify training adequacy. Additionally, facility representatives were assigned in-plant responsibilities and duties to assess the contractor shift crew performance of new or revised procedures and to assess the safety-consciousness reflected in the contractors day to day decision-making. Through these performance based inspections, the facility representatives have become increasingly knowledgeable in the specifics of the contractor procedures, conduct of operations instructions, and administrative policies and procedures. Additionally, they have become more knowledgeable of the affect that the cycle training had on the contractor's staff performance. This was considered to be of greater overall benefit than attendance in cycle training.

The Reactor Engineering Division (RED) specifies training requirements for its staff in SRSPo Manual Reactor Engineering Division Procedure (EP) EP-303, "Personnel Training," (Reference 27). As in the case of ROTD, the RED training requirements are also described in AI-106, which was published after EP-303. Records are maintained for the RED staff by the SRSPo Training Coordinator. Their training requirements include: (1) the four basic training courses (i.e., the courses all SRSPo employees take); (2) assessment techniques; (3) reactor technology; (4) system technology; (5) configuration management (including safety evaluations and unreviewed safety issues); (6) functional testing; (7) self-study, including selected DOE Orders, the Safety Analysis Report, selected INPO Good Practices and Guidelines, and selected NRC Inspection and Enforcement Manual chapters.

The training requirements for the RED staff, as documented in EP-303, meet the requirements of the restart criteria with the exception of training in the

duties, responsibilities, and authorities of staff positions and the training for lessons learned at SR or other facilities, which is now addressed in AI-106, Revision 1.

The only restart training required of the RED staff consists of the three courses designated by an asterisk (indicating they are mandated restart training for technical personnel) in Section 2.1.2(4)(b). With the exception of the duties, responsibilities, and authorities course, which has yet to be developed, essentially all of the RED staff has completed the other mandated restart training courses cited in Section 2.1.2(4)(b).

Training requirements for the Safety Oversight Division (SOD) and Technical Support Division (TSD) are documented exclusively in AI-106, Revision 1. These divisions also use the standard TRR. SOD staff training includes: (1) the four basic courses; (2) the four section 2.1.2(4)(b) courses; (3) guidance for conducting safety evaluations; (4) environmental awareness; (5) operating envelope safety; (6) introduction to ANSI/ASME NQA-1; (7) quality assurance, codes and standards; (8) simulator training; (9) special test programs; (10) system engineering concepts; (11) reactor physics and thermodynamics; (12) integrated start-up and shutdown; (13) introduction to component handling; (14) introduction to SR reactors; (15) INPO observations; (16) safety inspections; (17) appraising SR contractor performance; (18) duties, responsibilities, and authorities of staff technical positions; (19) fundamentals for DOE operations course; (20) Lessons Learned in the Operation of Nuclear Facilities; (21) Tiger Team methodology; (22) team building; (23) Management Oversight and Risk Tree (MORT); (24) MORT accident investigation techniques; (25) MORT-based root cause analysis; and (26) self study of the Safety Analysis Report, the reactor system, selected DOE Orders, portions of the Code of Federal Regulations, nuclear industry codes and standards, INPO Good Practices and Guidelines, sections and chapters of the NRC Inspection and Enforcement Manual, and DuPont standard operating procedures.

The training requirements for the SOD staff, as documented in AI-106, Revision 1, meet the requirements of the restart criteria. As indicated previously for the RED staff, essentially all of the SOD staff has completed the restart training mandated in Section 2.1.2(4)(b), with the exception of the yet-to-be developed duties, responsibilities, and authorities course.

TSD staff training includes: (1) the four basic courses; (2) the four section 2.1.2(4)(b) safety courses; (3) guidance for conducting a safety evaluation; (4) introduction to ANSI/ASME NQA-1; (5) quality assurance, codes and standards; (6) introduction to SR reactors; (7) INPO observation techniques; (8) safety inspections; (9) contract administration for technical representatives; (10) appraising SR contractor performance; (11) SRS project management; (12) duties, responsibilities, and authorities of staff technical positions; (13) Lessons Learned in the Operation of Nuclear Facilities; (14) a team building; and (15) self study of the Safety Analysis Report, selected DOE Orders, portions of the Code of Federal Regulations, nuclear industry codes and standards, INPO Good Practices and Guidelines. The training requirements for the TSD staff, as documented in draft AI-106, Revision 1, meet the requirements of the restart criteria.

A review and evaluation of the training discipline area indicates that one open item remains: SRSPo managers, facility representatives, and staff must complete their respective mandated restart training requirements.

Inspections and Audits

DOE will perform an Operational Readiness Review (ORR) of the SR reactors prior to the restart of each reactor. The purpose of the ORR is to provide final assurance that the reactors are ready to operate. The ORR Program Plan (as described in AI-107) and the actual review dates have not been formally approved. AI-107 also requires training for the ORR staff. This training, which is scheduled for early 1991, will include an overview of the ORR program, an introduction to SR reactor systems, human factors training, and a site-specific general employee training program. A dry run mini-ORR was conducted in July 1990 to increase familiarity with AI-107 and the reactors. The ORR will review a broad cross-section of issues, including management operations, quality assurance engineering, and human factors. AI-107 has been reviewed and is consistent with the restart criteria.

OG-450 and OG-451 provide formal guidance for the performance of inspections and surveillances of the SR reactors by the ROTD staff. These guidelines were modeled after the NRC's Inspection Plan. OG-450 includes guidance for periodic inspections to be performed on daily, monthly, and bimonthly schedules. It also provides guidance on performing risk-based inspections and on the minimum inspection requirements for the SR reactor facilities. The attachments to OG-450 and OG-451 provide specific guidance for a broad range of inspection and surveillance activities, including a requirement that inspectors be familiar with the inspection criteria before performing the inspection. The inspection guidance in OG-450 and OG-451, as implemented, enhances the safe operation of these facilities. The other three SRSPo divisions have comparable guidelines.

Each of the divisions issues a report containing the results of their periodic inspections from the previous month. This report, which is presented to the contractor, provides timely information on SRSPo findings. The SRSPo divisions also hold monthly meetings with the contractor to discuss and, if necessary, clarify findings and provide for contractor feedback on a real-time basis. All inspection findings are included in the SRSPo monthly report with unresolved findings (items not closed by the end of the month) also being entered into the SRSPo issue management data system.

Additional SRSPo inspection procedures/guidelines will be developed, as needed, to complement the SRSPo inspection programs discussed above. Upon completion, these procedures/guidelines will be evaluated and reported in a supplement to the SER. Those reviewed to date, including OG-450 and OG-451, were found to be consistent with the restart criteria. Based on the above, the inspections and audits criterion is acceptably satisfied.

2.1.4 Conclusion

SRSPo has made substantial progress to improve its oversight of the SR reactors. Some actions are not yet complete; primarily, the approval and

publication of all mandated restart documents (instructions, guidelines, procedures, etc.) and management and staff's completion of their restart training. Additional open items are clearly identified. A review of subsequent progress with respect to closing the remaining open items will be included in a supplement to the SER.

2.1.5 References

1. Secretary of Energy Notice 6-89 (SEN-6-89), dated May 19, 1989.
2. U.S. Department of Energy, "Savannah River Plant K-Reactor Restart Strategy," November 25, 1988.
3. Secretary of Energy Ten-Point Plan, announced June 27, 1989.
4. Letter, from P. W. Kasper, Manager of the Savannah River Operations Office, to James Moore, President of the Westinghouse Savannah River Company (WSRC), dated August 11, 1989.
5. Letter, from Steven D. Richardson, Director of the Office of Savannah River Restart to James Gallagher, Vice President and General Manager of the Reactor Restart Division of WSRC, dated March 15, 1990.
6. Memoranda, from Steven D. Richardson and Frank McCoy to the staffs of SRRO and SRSP0 and their support contractors, dated April 13, 1990.
7. Administrative Instruction 102, "SRSP0 Organization and Responsibilities," dated January 30, 1990.
8. Administrative Instruction 2, "Roles and Responsibilities."
9. Administrative Instruction 100, "Manual Organization," dated January 27, 1990.
10. Operating Guidelines 450, "Operations Inspections," dated March 16, 1990.
11. Operating Guidelines 451, "Maintenance and Surveillance Inspections," dated March 16, 1990.
12. Administrative Instruction 103, "Management Walkthroughs," Revision 1, dated December 18, 1990.
13. Administrative Instruction 104, "SRSP0 Notification to the Contractor of an Adverse Condition/Stop Work Order," dated January 30, 1990.
14. Administrative Instruction 105, "Event Notification and Investigation," dated December 14, 1990.
15. Administrative Instruction 107, "Operational Readiness Review," Revision 1, dated December 28, 1990.

16. Administrative Instruction 112, "Commitment Tracking and Document Tracking."
17. Administrative Instruction 119, "Issue Management," dated November 14, 1990.
18. Administrative Instruction 124, "Unreviewed Safety Questions."
19. Operating Guideline 457, "Preparation for Refueling," dated December 11, 1990.
20. Operating Guideline 458, "Refueling Activities," dated December 11, 1990.
21. Operating Guideline 409, "K-Reactor Restart Evolutions."
22. Operating Guideline 401, "Duty Officer," November 16, 1990.
23. Operating Guideline 404, "Event Notification and Reporting."
24. Administrative Instruction 121, "Safety Evaluation Report Assessment Program."
25. Administrative Instruction 106, "Training," Revision 1, dated February 8, 1991.
26. Operating Procedure 403, "Personnel Training," dated March 16, 1990.
27. Engineering Procedure 303, "Personnel Training," dated March 27, 1990.

2.2 Management and Safety Culture

2.2.1 Background

During the attempted start up of P-Reactor in August, 1988, an apparent reactivity anomaly occurred. This anomaly resulted from the failure to account for the "poison" that had accumulated in the reactor core while the facility was shut down. The lack of concern displayed by the operators during the attempted startup, along with the failure to notify management of both the previous operating contractor and DOE, raised serious questions as to the ability of the previous operating contractor's staff to safely operate the reactors. DOE suspended operations at all of its production reactors until identified weaknesses could be demonstrated to be resolved. These weaknesses included non-conservative management decisions, inadequate procedures, and training. All three reactors have been maintained in a shutdown condition since the August 1988 incident, pending the results of an in-depth review of management and operational problems at the SRS reactors and the completion of modifications necessary to enhance safety.

The commercial nuclear reactor accident at Three Mile Island (TMI) Unit 2 resulted in a number of hardware upgrades and a significant change in the management of operations at commercial nuclear plants regulated by the Nuclear Regulatory Commission (NRC). Most particularly, the commercial nuclear industry came to a growing understanding that hardware alone could not be relied upon for nuclear safety. A new discipline of operations and a rising standard of excellence, largely based on those already in place in the U.S. Naval nuclear program, were put into place in the commercial nuclear industry.

In the area of conduct of operations at SRS, production and industrial safety (the day-to-day protection of employees from occupational hazards of the facilities) were emphasized over reactor safety (the minimization of the probability and consequences of potential reactor accidents that might otherwise pose undue risk to the public). This operating style, which was encouraged by DOE, along with the lack of interaction with the commercial nuclear power community and only limited review from offsite organizations, led to the creation of a vastly different operating mentality between the two communities. There was also a prevailing attitude that the operation of light water reactors held little or no applicability to heavy water reactor operations. The safety culture development that occurred within the commercial and naval nuclear complexes was, therefore, lacking at the DOE facilities. This was further aggravated by the lack of an aggressive safety posture by DOE.

2.2.2 Restart Criteria

(1) Safety Culture

The contractor shall have a written, formally implemented safety-first policy in the form of administrative procedures which:

- o place reactor safety above all other considerations

- o elevate management support of and overall peer status of reactor operations personnel by recognizing the special skills, knowledge, and responsibilities operators are required to have
- o clearly define job responsibilities and authority of all personnel in reactor operations
- o establish rules for professional conduct
- o provide operational performance standards
- o institute a self-assessment process to ensure continued improvement

(2) Organization and Line Accountability

- (a) In order to support a positive safety-management culture, the contractor's reactor operations organization shall have well-defined and understood lines of authority. Specifically, the contractor shall provide administrative procedures that document the structure of the reactor operations organization, including job descriptions, authorities, responsibilities, and interfaces with support groups.
- (b) The contractor shall create a plant manager position for each reactor. The plant manager shall be responsible for all aspects of reactor operation, including:
 - o reactor safety
 - o compliance with DOE Orders
 - o maintenance and engineering direction
 - o radiological controls
 - o power (electrical and water)
 - o housekeeping
 - o emergency preparedness
 - o startup authority

(3) Professionalism in Performance of Plant Activities

- (a) Professionalism among both management and operations staff personnel, which encourages professional conduct in the working environment, is the most important aspect of developing a safety culture attitude. The contractor shall foster development of professionalism throughout all levels within the organization by

actively supporting the following qualities in operating personnel:

- o demonstrated personal integrity and honesty with formal behavior, conduct, and appearance in all job tasks
 - o commitment to excellence
 - o highly trained and competent staff with continuous effort to maintain and improve job-related knowledge and skills
 - o willingly accepted responsibility for all actions and an attitude of ownership of the plant
 - o operation of the plant in a manner which demonstrates that all decisions place public and reactor safety considerations first
 - o maintenance of a plant awareness and continuous monitoring and response to all plant conditions
 - o strict compliance with procedures to ensure public protection and plant safety
- (b) The contractor shall promote a professional working environment and develop and maintain administrative procedures which clearly define operating standards of conduct including: control room access/decorum; investigation of abnormal indications/events; logkeeping; shift turnover; housekeeping; fitness for duty; and radiological protection practices. Contractor employee performance evaluations shall be made based on nuclear safety dedication, as well as performance standards related to the maintenance of a positive safety culture attitude.

(4) Communications

The contractor shall communicate goals, standards, and policies by a system specified by appropriate administrative procedures. Communication methods to be considered should include: standing orders and instructions; night orders; plans of the day; status reports. The contractor shall encourage an atmosphere conducive to constructive criticism and feedback, with management response communicated back to plant personnel in order to encourage continued input.

(5) Management

- (a) Management Direction: Contractor management shall have primary responsibility for fostering a safety culture environment by the way it directs its organization. Specifically, the contractor's management staff shall:

- o Set the example by conducting itself with commitment and integrity.
 - o Maintain an atmosphere of open communications such that problems are brought to its attention undiluted.
 - o Set goals that encourage continual improvement in performance and avoid a sense of self-satisfaction or complacency.
 - o Be proactive and responsive, pursuing solutions to problems with the objective of correcting root causes and improving performance.
 - o Establish a training program to develop and enhance safety culture throughout all management levels.
- (b) Management Involvement: The contractor's management shall have both proceduralized and informal methods of staying involved with the day-to-day reactor plant activities. As a minimum, management monitoring of activities shall include observation and assessment of:
- o procedure compliance
 - o operator proficiency and competence in job tasks
 - o plant status(e.g., operability of equipment/instrument/annunciator and their impact on operations job performance and overall safety
 - o tagouts/lockouts
 - o housekeeping
 - o operator attitudes
- (c) Management involvement through formal assessments of plant activities shall be conducted in accordance with established procedures, with the findings entered into a formal tracking system. These findings should be used to facilitate early identification and correction of problem areas. Management assessments shall routinely include backshifts, weekends, and holidays, and shall be rotated among all levels of management.

WSRC shall also meet the following commitments post-restart.

Employee Concern Program

A formal administrative procedure shall be in place to address and provide feedback on the resolution of employee concerns. This procedure shall also address a course of action for dispositioning concerns which cannot be

resolved to an employee's satisfaction. The procedure should clearly state that an employee's pursuit of issues believed to be related to reactor safety will in no way subject the individual to any punitive action.

2.2.3 Evaluation

A large number of management, staffing, and organizational changes have taken place since the SRS reactors began their current extended outages. The most significant change in this area occurred on April 1, 1989, when the operating contract for SRS was transferred from the previous operating contractor to the Westinghouse Savannah River Company (WSRC).

Westinghouse has long been a leader in the nuclear industry at both the national and international levels. The corporate goals reflect the company commitment to maintain this important leadership position as well as to efficiently and safely operate the SRS production reactors.

The President of WSRC reports to the Executive Vice President, Energy and Utility Systems, who in turn reports to the President and Chief Operating Officer of Westinghouse. This placement of the WSRC within the Westinghouse corporate reporting structure indicates the high priority given SRS and ensures the direct involvement of senior Westinghouse corporate management.

In order to familiarize itself with WSRC programs and initiatives in the area of management and safety culture, the DOE staff reviewed WSRC-RP-90-52, dated January 19, 1990 (Reference 1), prior to starting a detailed evaluation. Reference 1 is a White Paper concerning the Reactor Restart Division culture change program. In this document, WSRC states that the White Paper and its attachments provide the integrated culture change program being implemented for the SRS reactor program. The White Paper describes organizational changes and WSRC goals and initiatives and includes internal correspondence, newsletters, etc. This documentation provides clear evidence that WSRC senior management recognizes the importance of a safety culture and is proactively taking the initiative to institutionalize the required culture.

Since the starting point for establishing a safety culture is with the senior management, it is the opinion of the DOE staff that the change in operating contractors subsequent to the start of the current reactor outages is a major step towards achieving the management and safety culture desired and expected by the DOE. The new operating contractor, WSRC, brings the more safety-conscious philosophy present in the commercial nuclear industry to SRS. Most of the senior management now in place come from within Westinghouse and have come to SRS since April 1989.

Since assuming responsibility for SRS, WSRC has also brought in senior managers and many new line managers with previous experience in the commercial nuclear industry and has significantly augmented the professional staff with people who have commercial or naval nuclear experience.

The management and staffing changes described above clearly provide the environment for the management and safety culture changes that are necessary at SRS. The new senior management must continue to set the example as role

models and aggressively provide the direction and guidance required to ensure that the attitudes, behavior, and values of a safety culture are well understood, accepted, and practiced. The new professional staff must also continue to use every opportunity to act as agents of change for the improved way of conducting business.

In order to assess compliance with the restart criteria and its implementation, a team consisting of DOE staff and consultants conducted numerous interviews of WSRC management and staff; performed walkthroughs at K-Reactor; and reviewed documents, such as administrative procedures, policies, newsletters, and memoranda.

During the period of May 14-24, 1990, the evaluation team conducted 70 formal interviews. Personnel interviewed included the President of WSRC and virtually all top and mid-level managers in the Reactor Restart Division (RRD). Several managers and staff members in the Reactor Operations Department (ROD) were interviewed, including the K-Plant Manager; Central Control Room Operators (SROs); Building Operators (ROs); and managers and staff in Reactor Technical Engineering, Health Protection (HP), and Mechanical and Electrical & Instrumentation (E&I) Maintenance. Each interviewee was questioned by at least two evaluation team members in order to assess compliance with the acceptance criteria, particularly in the area of safety culture (attitudes and beliefs and the communication and reinforcement of that culture).

In January 1991, an evaluation team interviewed Paul Rice, Vice President and General Manager, RRD, and Steve Smith, Manager, ROD. These senior managers joined WSRC in November 1990. Based on the results of the interviews and reviews of various documentation, the DOE staff's evaluation of the restart criteria is given below.

Safety Culture

WSRC has identified a policy on nuclear facility safety in its Conduct of Operations Manual (COM), Rev. 0, (Reference 2), which was formally implemented May 22, 1990. The Introduction section of this document states that "as the operator of a nuclear facility you are entrusted with certain specific responsibilities. Under all circumstances, the safety of the reactor must be your primary consideration."

Based on the above, WSRC has provided a written, formally implemented, safety-first policy in the form of an administrative procedure that places reactor safety above all other considerations. However, WSRC should use the monthly safety meetings as a forum to communicate to all employees that reactor safety comes before all other considerations. Pending the above, acceptable satisfaction of this criterion remains an open item.

The special skills, knowledge, and responsibilities operators are required to have are well recognized by WSRC, as evidenced by the extensive amount of training (discussed elsewhere in this report) that operators are required to successfully complete in order to be certified. Additionally, Section 3 of

the COM defines the responsibilities of operators while on shift. That section states, in part, that "The conduct of safe operation requires the thoughtful, skillful application of individual knowledge and experience." The evaluation team was also told by the operators it interviewed that, in general, the operators felt their status has been enhanced. Based on the above, we find that the criterion to have a written, formally implemented, safety-first policy in the form of administrative procedures that elevate management support of, and overall peer status of, reactor operations personnel by recognizing the special skills, knowledge, and responsibilities operators are required to have is acceptably satisfied.

The staff interviewed operations personnel and reviewed department organization manuals to determine whether all personnel in reactor operations had clearly defined job responsibilities and authority. Sections 3 and 3A of the COM define operator shift responsibilities and shift management responsibilities and authority. Reactor Operations Standing Instruction (ROSI) III.1 (Reference 3) provides the organization of reactor operations. It defines the authorities and responsibilities of reactor department personnel including: Operations Managers, Building and Central Control Room Supervisors, Reactor Building and Central Control Room Operators, and Shift Managers. ROSI III.1 does not adequately address the Plant Manager's responsibilities, because it does not state that the Plant Manager shall be responsible for all aspects of reactor operation, including those identified in the restart criteria.

Based on the above, the staff finds that the criterion to clearly define job responsibilities and authorities for ROD personnel is a confirmatory item pending full inclusion of the Plant Manager's responsibilities.

Rules for professional conduct are contained in the COM, Section 2, Part B, "Professional Behavior." Items addressed include formality of activities, distractions, work environment, and courtesy. The evaluation team observed personnel in the Central Control Room and found they were conducting their activities in a professional manner. Based on the above, we find that the criterion to establish rules for professional conduct is acceptably satisfied.

Operational performance standards are found primarily in the Guiding Principles of Reactor Operations (Reference 4), commonly referred to as the five guiding principles. These principles, which address mission, ownership, procedural compliance, self-assessment, and motivation, are displayed in virtually every ROD manager's office and in other pertinent places and have been distributed to ROD personnel and discussed with them by management for purposes of reinforcing the principles.

The evaluation team found that the employees interviewed in reactor operations were aware of the five guiding principles; while they could not quote them verbatim, in general, they understood and believed in the intent of the principles. RRD has a similar set of principles, which address mission, vision, and the quality-first concept, that was communicated to senior and mid-level management and to their staffs, as evidenced during the evaluation team's interviews. Health Protection (HP) personnel have their own, similar principles of excellence, which are used to evaluate performance.

Additionally, the reactor operators developed a code of ethics (Reference 5), adopted at all three reactor facilities, that commits them to operate those facilities in a safe, reliable, and skilled manner. The operators interviewed by the evaluation team believed in and embraced that code of ethics. Based on the above, we find that the criterion to provide operational performance standards is acceptably satisfied.

Performance of the ROD is being assessed monthly, using several performance indicators for each of the five guiding principles. Various Institute of Nuclear Power Operations (INPO) and commercial industry performance indicators are used to enable objective monitoring of personnel and equipment performance for comparison against a set of stated goals.

The April 1990 monthly performance indicator report (Reference 6) provides a good example of the results of this self-assessment process. Other self-assessments have been or are being performed. For example, a self-assessment in the area of reactor maintenance (Reference 7) was conducted during the period of March 1989 through January 1990. The criteria used for the self-assessment process were based on INPO guidelines. The assessment results were documented, then used to develop a reactor maintenance improvement plan (Reference 8). The plan is intended to be implemented over a 3-year period to improve maintenance at the site. Based on the above, we find that the criterion to institute a self-assessment process to ensure continued improvement is acceptably satisfied.

Organization and Line Accountability

The structure of ROD is documented in Attachment 3 of ROSI III.1 (Reference 3). Revision 5 of ROSI III.1 reflects the current ROD organization and addresses the responsibilities and authorities of all ROD personnel and their interfaces with support groups. Based on the above, the criterion for the operating contractor to provide administrative procedures that document the structure of the reactor operations organization, including job descriptions, authorities, responsibilities, and interfaces with support groups is acceptably satisfied.

WSRC has created a plant manager position for each reactor. The latest effective revision of ROSI III.1 identifies the position of Plant Manager; however, it is not documented that the Plant Manager is responsible for all aspects of reactor operation, including those identified in the restart criteria. The COM is being revised and will incorporate DPSOP 336 (ROSI) in an effort to eliminate conflicting or duplicate requirements. Currently, the COM does not adequately address the Plant Manager's responsibilities. Reference 9 discusses plant manager responsibilities in the 100's areas. Specifically, it addresses the ROD takeover of control of the 190/186 buildings from the Power Operations Department and outlines a plan for future control in the 100's areas. Also, during discussions, top managers have expressed concern over the matrixing of the HP area. The current matrix does not include HP under ROD. As an interim measure, the Manager, ROD has instituted an HP ALARA (As Low as Reasonably Achievable) "Tiger Team" within ROD. Based on the above, the criterion for the contractor to create a plant manager position for each reactor is acceptably satisfied. However, the

criterion that the plant manager shall be responsible for all aspects of reactor operation, including those identified in the criterion, remains an open item.

Professionalism in Performance of Plant Activities

Personal integrity and honesty, with formal behavior, conduct, and appearance for all operating personnel, are addressed in the COM. Formal behavior and conduct are addressed in Section 2, Part B, of the COM, and are discussed above. Rules concerning appearance are contained in Part C of Section 2. Issues addressed include personal appearance, cleanliness, and uniforms. Operators interviewed and observed in the central control room by the evaluation team were wearing the uniforms provided to them. Personal integrity and honesty is addressed in Part F of Section 2 of the COM. When addressing the reliability and accountability of operators, the COM states, in part, that "Personal reliability requires integrity. An individual's statements must be truthful." Additionally, the reactor operator's code of ethics, previously discussed, states, in part, that "we commit ourselves to perform our duties with the highest personal standards of honesty, integrity, and ethics achievable." Based on the above, we find that the criterion for WSRC to actively support the qualities in operating personnel of demonstrated personal integrity and honesty with formal behavior, conduct, and appearance in all job tasks is acceptably satisfied.

Section 2, Part A of the COM contains a commitment to excellence. It states, in part, that "The Managers and Operators of the Reactor Operations Department are committed to excellence in the performance of nuclear operations." The reactor operator's code of ethics, strongly supported by WSRC management, also contains a commitment to strive for excellence in operational safety and "To maintain the highest standards of excellence in human performance" The operators interviewed by the evaluation team clearly indicated a strong desire to carry out their duties in accordance with this commitment. Based on the above, we find the criterion for WSRC to actively support the quality in operating personnel of a commitment to excellence is acceptably satisfied.

Extensive and continuing training, as discussed elsewhere in this SER, is required for operating personnel to provide and improve job-related knowledge and skills. A significant amount of training is now required for Central Control Room Operators (SROs) in order to be certified, and the certification must be maintained current. In accordance with Section 3, Part A, of the COM "Only a Certified Central Control Room Operator, or Supervisor holding certification, may manipulate the controls of the reactor...." Building Operators (ROs) are also required to be sufficiently trained. The operators interviewed by the evaluation team (1) commented about the extensive amount of training they are now required to receive since WSRC became the operating contractor, (2) believed the training was excellent for providing the knowledge and skills they needed, and (3) felt much more prepared and comfortable when performing their responsibilities. The reactor operator's Code of Ethics also commits the operators to perform their duties in a competent, knowledgeable, and skilled manner. Based on the above, we find that the criterion for WSRC to actively support the qualities in

operating personnel to be a highly trained and competent staff with continuous effort to maintain and improve job-related knowledge and skills is acceptably satisfied.

Accepting responsibility for all actions and an attitude of ownership of the plant are actively promoted and supported by the operating contractor. For example, Section 2, Part F, of the COM states, in part, that "Operations personnel must have the capability and confidence to be accountable for their actions. Being accountable means that the operator is willing to assume the responsibilities for the position to which the individual is assigned...." Part B of Section 2A of the COM states, in part, that "Shift Management shall hold Operators and personnel from other groups assigned to support Operations accountable for their performance. This will be accomplished through management counseling, performance appraisal, and when necessary, disciplinary action." The reactor operator's Code of Ethics commits the operators "To exemplify responsibility, teamwork, accountability and effective leadership." The second of the five guiding principles fosters ownership. That principle states "We own these plants. Nothing happens in our plants without our forethought, our knowledge, and our approval. If we don't approve it, it doesn't get done. If we approve it and it's wrong, it's our fault--not someone else's."

The operators interviewed by the evaluation team appeared willing to accept responsibility for their actions and understood they were the owners of their plant, not the custodians. Based on the above, we find that the criterion for WSRC to actively support the qualities in operating personnel to willingly accept responsibility for all actions and an attitude of ownership of the plant is acceptably satisfied.

Section 1, Part A, of the COM states, in part, that "It is the policy of Reactor Operations, that the primary consideration in the operation of the Company's facilities is the safety of the general public, station personnel, and equipment. All operations personnel, when faced with any abnormal plant condition, transient, or event, are to take the safe course of action. Completion of scheduled activities is secondary." The reactor operator's Code of Ethics commits "To acknowledge that safety outweighs all other considerations including production, economics, and the expediency of reactor operations." The operators interviewed by the evaluation team are serious about their Code of Ethics and about protecting the public. Based on the above, we find that the criterion for WSRC to actively support the quality in operating personnel to operate the plant in a manner which demonstrates that all decisions place public and reactor safety considerations first is acceptably satisfied.

Maintaining a plant awareness and monitoring and responding to plant conditions are addressed in several places in the COM. For example, Section 2, Part A, of the COM states, in part, that Reactor Operations personnel should "Constantly remain alert and maintain awareness of plant status, anticipating conditions that could adversely affect plant reliability." Part B of Section 2 states, in part, that Shift Operations personnel shall "Be attentive to system conditions and parameter trends by monitoring the information displays associated with each person's duties." Part E of Section

2 states, in part, that "Operators shall take prompt actions to determine the cause of the abnormalities, advise Shift Management of the abnormalities, and take corrective action." Section 7, Parts A and B, also provide guidance regarding response to control room alarms and indications. The reactor operator's Code of Ethics commits the operators "To always be attentive, observant, and knowledgeable of the system status with respect to our equipment and facilities."

Based on the above, we find that the criterion for WSRC to actively support the qualities in operating personnel to maintain a plant awareness and continuous monitoring and response to all plant conditions is acceptably satisfied.

Various programs, procedures or requirements exist for operating standards of conduct, such as control room access/decorum, investigation of abnormal indications/events, logkeeping, shift turnover, housekeeping, fitness for duty, and radiological protection practices. Except for investigation of abnormal indications/events, all the above are discussed and evaluated elsewhere in this report. To varying degrees, the COM addresses the above and many other operating standards of conduct. For example, Section 2, Part E, requires operators to take prompt actions to determine the cause of alarm abnormalities, advise shift management, and take corrective action. Part B of Section 7 of the COM requires that, if an indication is suspected of being inaccurate, operations personnel "Initiate an investigation of the device displaying the unusual or abnormal information." Part D of Section 13 discusses the Operations Manager's monthly report, which provides a summary of significant operational events investigated. Section 14 of the COM describes actions to investigate and resolve incidents and near-miss situations. Control room access requirements are described in Section 5, Part G of the COM. Logkeeping requirements are addressed in Section 8. Part E of Section 6 of the COM discusses the shift turnover process. Section 8, Part F describes guidelines for Reactor Building Operator's inspections, which include satisfactory housekeeping and cleanliness and having clearly marked radiation and contamination areas. Part C of Section 13 addresses management housekeeping walkdowns. Fitness for duty requirements are described in Section 2, Part D, of the COM.

Based on the above, we find that the criterion for the contractor to develop and maintain administrative procedures that clearly define operating standards of conduct, including those cited above, is acceptably satisfied.

Procedural compliance is one of the five guiding principles. That principle states "we follow our procedures religiously, but not stupidly. If the procedure is wrong, we fix it. If it can be improved, we improve it." Section 9 of the COM addresses use of procedures. Part C states, in part, that "Operations shall be conducted in accordance with approved Operating Procedures." Proper use of procedures, and emphasis on how important that is, is included in the operators' training program, discussed elsewhere in this SER. In their Code of Ethics, the reactor operators commit "To diligently apply our policy and procedures, devoting ourselves to being professionals with impeccable standards."

Based on the above, we find that the criterion for WSRC to actively support the quality in operating personnel regarding strict compliance with procedures to ensure public protection and plant safety is acceptably satisfied.

Section 2, Part B, of the COM addresses professional behavior, including performing only activities essential to supporting safe plant operation, computer usage, courtesy, and professional attitude. Part C of Section 2 of the COM addresses appearance, including the use of uniforms. Section 7, Part J, of the COM addresses professional reading material to be read by shift operations personnel. By their Code of Ethics, the reactor operators commit themselves to being professionals. Based on the above, we find that the criterion for the contractor to promote a professional working environment is acceptably satisfied.

With regard to employee performance evaluations, WSRC should ensure that performance evaluations for appropriate employees are based on nuclear safety dedication and include performance standards related to the maintenance of a positive safety culture attitude. Pending the above, acceptable satisfaction of this criterion remains an open item.

Communications

Communication of goals, standards, and policies is accomplished by several methods. Communication methods include standing orders and instructions, night orders, plans of the day, and status reports. Section 4, Part A, of the COM lists, as examples of formal documents used in Operations, technical specifications, area standing instructions, reactor operations standing instructions, shift orders, narrative logs, shift turnover checksheets, data sheets, and tags. Goals for performance indicators associated with the five guiding principals are contained in the monthly performance indicator report. Standards, such as the five guiding principles, are contained in the COM, and have also been communicated by WSRC interoffice memorandum (Reference 10) and posted in pertinent places, including managers offices. Standards for conduct of operation are contained in such documents as reactor department standing instructions (ROSIs) and the COM. Policies are disseminated by WSRC policy manuals, and can also be found in the COM. Other parts of the overall written communications program are newsletters and bulletin boards. Oral communication methods include videotapes, roundtables, culture change seminars, and rallies.

Based on the above, the staff finds that the criterion for the contractor to communicate goals, standards, and policies by a system specified by appropriate administrative procedures is acceptably satisfied.

An open communication channel with all employees was encouraged through the "Tell It To Jim" campaign. This program allowed any employee to submit written complaints, comments, questions, or concerns directly to Jim Gallagher, the former Vice President and General Manager, RRD. The program provided employees with ready accessibility to senior management and provided senior management another opportunity to communicate and reinforce aspects of the culture change.

In November, 1990, Paul Rice took over as Vice President and General Manager, Reactor Restart Division. His 27 years in nuclear engineering and management include naval and commercial nuclear power experience and membership on the Advisory Committee for Nuclear Facility Safety (ACNFS). He brings to WSRC a strong safety-first philosophy. Through a "hands-on" management approach, Paul Rice continues to provide high-level management attention and feedback to reactor plant areas of concern. During that transition, Steve Smith took over as Manager, Reactor Operations Department. He also brings an aggressive safety-first philosophy, supported by 22 years of experience in both commercial and naval nuclear power. During interviews, both men expressed concern over the efficiency of communications and control of plant activities. They identified a plan to alter management structure in an effort to streamline operations and improve communications and program implementation. The proposed reorganization consists of: General Manager, RRD; Manager, ROD; Plant Manager, Operations Manager, and Shift Manager.

Constructive criticism and feedback is also provided to employees through the critiques that are performed of plant incidents. These critiques provide valuable lessons learned, and interviews by the evaluation team confirmed that they are required reading for operators.

The "Pats on the Back" program is another mechanism by which management encourages and highlights desirable employee performance and input and commends them, in writing, for their efforts. (See Reference 11, the May 11, 1990, "Pats on the Back" Weekly Report for an example of this program.)

Based on the above, the staff finds that the criterion for the contractor to encourage an atmosphere conducive to constructive criticism and feedback, with management response communicated back to plant personnel in order to encourage continued input is acceptably satisfied.

Management

Management Direction

Commitment, and more so, integrity are not easily measured qualities. Many of the people interviewed by the evaluation team were senior or mid-level managers. Their responses to questions and the discussions that took place during those interviews are indicative of people with integrity and people who are strongly committed to performing a quality job to assure the timely, safe restart and operation of the reactors. Based on the above, we find that the criterion for the contractor's management staff to set the example by conducting itself with commitment and integrity is acceptably satisfied.

An atmosphere of open communications appears to exist at SRS. Activities such as roundtables, restart rallies, and management visits to operation training classes are all mechanisms that provide open communication between employees and management. Comments made by personnel interviewed by the evaluation team included the comments that "you can speak to the Plant Manager" and that "he has an open door policy." Based on the above, we find that the criterion for

the contractor's management staff to maintain an atmosphere of open communications such that problems are brought to its attention undiluted is acceptably satisfied.

The primary mechanism being used to encourage improved performance and to avoid a sense of self-satisfaction or complacency is the monthly ROD performance assessment, previously discussed. Using various INPO and commercial industry performance indicators (each of which are, then, related to one of the five guiding principles), personnel and equipment performance are monitored and compared against a set of goals. Based on the above, we find that the criterion for the contractor's management staff to set goals that encourage continual improvement in performance and avoid a sense of self-satisfaction or complacency is acceptably satisfied.

There is evidence that WSRC is being proactive and responsive, pursuing solutions to problems with the objective of correcting root causes and improving performance. Critiques, which are one example of this evidence, are a review by operations management of all known information about the subject incident. The focus is on the determination of suspected root causes and assignment of additional action items to determine the actual root causes so they can be corrected. The monthly ROD performance assessments identify performance areas needing improvement. Other self-assessments previously discussed, such as the detailed assessment of reactor maintenance, are being used to improve performance. Based on the above, we find that the criterion for the contractor's management staff to be proactive and responsive, pursuing solutions to problems with the objective of correcting root causes and improving performance is acceptably satisfied.

In general, the managers interviewed by the evaluation team were aware of the initiatives and the need to develop and enhance safety culture. They had been spoken to, and had talked to their people about the importance of safety, about training, and about the new way of doing business that includes the necessity for complete documentation, up-to-date drawings, quality assurance, and a disciplined approach to performing activities. They were also aware of and knowledgeable of the total quality concept, of doing things right the first time. They had the RRD mission, vision, and principles, or the ROD five guiding principles hanging on their office walls. Some of the managers interviewed had attended safety culture seminars, while others were new to the site, bringing with them the desired commercial and navy nuclear experience and safety philosophy. While there is no formal training program specifically for developing and enhancing safety culture throughout all management levels, in general the managers interviewed had clearly received the message. Therefore, in lieu of a formal training program an acceptable alternative would be to emphasize reactor safety and reinforce safety culture during the monthly safety meetings, which currently address industrial safety. Based on the above, we find that the criterion for the contractor's management staff to establish a training program to develop and enhance safety culture throughout all management levels is an open item, pending inclusion of discussions on reactor safety and safety culture in the monthly safety meetings.

Management Involvement

Involvement of management in day-to-day activities is required by various procedures. WSRC Procedure RDP 9.01, Rev. 1, entitled "Management Monitoring," with an effective date of August 28, 1990 (Reference 12), defines requirements and provides instructions for senior managers to conduct tours of the reactor plants and supporting facilities, with the focus of the tours being on those factors that directly or indirectly affect the safety or function of the facility or the successful accomplishment of the activity, project or program. Attachment A of the procedure identifies typical activities to be toured, including work control, testing, shift turnover, and maintenance. Attachment B identifies factors to be examined during tour, as applicable, including procedure compliance (both operations and maintenance), knowledge and control of operating status, locking and tagging, housekeeping, technical inquisitiveness, and adherence to safe working, radiation protection, and fire protection/mitigation practices. The senior managers identified as participating in the monitoring program include the Deputy General Manager, Reactor Operations; the Manager of Reactor Operations; the Station Manager; all three Plant Managers; and 19 other senior managers.

K-Area Standing Instruction (KSI) No. D-50, Rev. 0, entitled "K- Area Management Inspections," dated February 13, 1990 (Reference 13), defines requirements and provides instructions for managers to conduct inspections of the reactor facilities. Attachment A of the instruction identifies topics of inspections, including logkeeping, procedure documentation audit, procedure compliance field observation, watchstanding practices, communications, and HP and maintenance practices. Attachment B identifies factors to be examined during the inspection, as applicable, including procedure compliance, knowledge and control of operating status, locking and tagging, housekeeping, inquisitiveness, goal awareness, and adherence to safe, radiation protection, and fire protection/mitigation practices. The managers identified as having inspection responsibilities include the Plant Manager, the Operations Manager, the Maintenance Manager, the HP Manager, and several other managers.

There is also a Management Walkdown Program, dated March 27, 1990, for material condition in the 100-K Area (Reference 14). In addition to identifying deficiencies in plant material condition that must be corrected prior to restart, housekeeping items are to be identified, and radiation protection, occupational health, security, fire protection, personal safety, and environmental protection items are to be addressed. Where possible, deficiencies are to be tagged to assist personnel in performing corrective action. Many of the program's housekeeping requirements are being incorporated into ROD Housekeeping/Material Condition Control Procedure, RDP 3.02 (Reference 15). The procedure incorporates INPO-85-038 guidance and includes implementation, training, inspection standards, reporting and follow-up.

A Management Assessment Plan, dated December 14, 1989 (Reference 16), describes the program for management assessment of operators to determine the effectiveness of the training program. The assessments are conducted by observing operating crew performance during simulator exercises and in-plant

drills, i.e., operator proficiency and competence. The assessment team consists of senior managers from RRD, including the Vice President and General Manager.

Section 13 of the COM addresses operations oversight activities, performed by Operations Management personnel for the purpose of assessing operations performance. In general, these activities provide a mechanism for reviewing the technical knowledge, interpersonal relations, attitudes, supervisory performance, and personal characteristics of operations personnel by observation of on-the-job performance. The management oversight activities consist of observations, evaluations, housekeeping walkdowns, and monthly reports. Part A of Section 13 of the COM identifies attitudes displayed by the participants that should be included in the recorded facts of an observation. Based on the above, we find that the criterion for management monitoring of activities to include observation and assessment of the areas identified in the criterion is acceptably satisfied.

Procedures exist, as discussed above, for conducting formal management assessments of plant activities. Each one requires that results and findings be documented on designated forms. RDP 9.01 requires that significant deficiencies be entered into the appropriate tracking system for disposition. Area management is responsible for taking action in response to tour results. KSI D-50 requires that a unique tracking number be assigned to the inspection form by the Procedure Coordinator and that he/she be notified when corrective action is taken/completed. The material condition Management Walkdown Program requires that each deficiency be assigned an ID number for tracking purposes and states that a clerk/typist will be assigned to track deficiencies. The Management Assessment Plan states that assessors will be responsible for returning completed check sheets and associated comments to the Manager of Operations Assessment. The data and comments are to be compiled and used as an indicator of crew performance and for measuring progress. Section 13 of the COM requires that the Plant Manager and others meet at least monthly to review the collected observations, identify action items to correct problems and assign responsibility, and document the results of this evaluation.

Based on the above, we find that the criterion for management involvement through formal assessments of plant activities to be conducted in accordance with established procedures, with the findings entered into a formal tracking system, is acceptably satisfied.

As described above, the findings from the various management involvement activities are to be used to facilitate early identification and correction of problem areas. Therefore, we find that the criterion for these findings to be used to facilitate early identification and correction of problem areas is acceptably satisfied.

RDP 9.01 requires that tours by various senior managers, on a rotating schedule, be conducted on all shifts and during weekends and holidays. KSI D-50 requires the managers to conduct the inspections on back shifts and weekends. Section 13 of the COM requires that, in the course of each calendar quarter, a representative number of day shift, back shift, weekend or holiday activities inside and outside the Control Room be included in the management

observations. Based on the above, we find that the criterion for management assessments to routinely include backshifts, weekends, and holidays, and to be rotated among all levels of management is acceptably satisfied.

Employee Concerns Program

WSRC has established a policy regarding open communications of employee concerns in Management Policy (MP) 1.11 (Reference 17). This policy states, in part, that employees shall have the right and responsibility to report concerns, to receive a timely investigation and resolution of the concern, and protection from reprisal, reprimand, harassment, intimidation, retaliation, or criticism as a result of reporting the concern.

WSRC has established a procedure for reporting, documenting, validating, and responding to employee concerns in Management Requirements and Procedure (MRP) 1.06 (Reference 18). The procedure states, in part, that "all concerns may be stated without fear of discrimination or retribution." Furthermore, if the employee considers the response unsatisfactory, or if the employee believes that he/she has been subjected to harassment, reprisal, or intimidation as a result of expressing a concern, the employee may request a meeting with higher levels of management within the employee's division/department or with representatives from the ESH&QA and Human Resources Divisions. If the response continues to be unsatisfactory to the employee, he/she may continue to elevate the concern to higher levels of management and ultimately to the office of the President, whose decision on resolution of the concern shall be final.

MRP 1.06 provides detailed instruction for handling employee concerns. The screening process begins when a concern is identified. The concern undergoes an initial review for reportability and immediate actions under the direction of the Site Issues & Employee Concerns Coordinator. The concern is assigned a number, entered into a tracking system, and then assigned for investigation. Following the investigation, recommendations and actions are defined, and the employee is contacted for review and closure.

Employee concerns programmatic information is widely disseminated. Program posters are displayed on controlled bulletin boards throughout the site. An information video is shown during general employee training (GET), and employees are required to read a program overview and sign an acknowledgement form. WSRC management routinely provides program information through sitewide letters and mailings to all employees. Additionally, articles providing program information appear regularly in the SRS News.

In February, 1990 WSRC conducted a survey on the effectiveness of the Employee Concerns Program. An Action Plan to address program deficiencies was issued March 30, 1990 (Reference 19). The plan addresses three objectives: effective advertising, policy and procedure review, and development of performance indicators. Implementation of the plan is nearly complete. Closing tasks include issuance of a revision to MRP 1.06 and the evaluation of performance indicators, including an employee survey.

Based on the above, the post-restart criterion to establish a formal administrative procedure to address employee concerns, which not only discusses the dispositioning of concerns that were not resolved to the employee's satisfaction, but also states that the employee will in no way be subject to any punitive action, is acceptably satisfied.

2.2.4 Conclusion

Based on the above evaluation, the DOE staff concludes that WSRC has made substantial and acceptable progress toward satisfying the restart criteria for Management and Safety Culture. Issues remaining to be addressed are clearly identified in the evaluation. The DOE staff will update its evaluation and document the results in an SER Supplement.

2.2.5 References

1. WSRC-RP-90-52, "White Paper Reactor Restart Division Culture Change Program (U)," January 19, 1990.
2. WSRC, Conduct of Operations Manual, Rev. 0, May 22, 1990.
3. DuPont, Standard Operating Procedure (DPSOP) 336, Reactor Operations Standing Instruction (ROSI) III.1, "Organization of the Reactor Operations Department," Rev. 5, July 15, 1990.
4. WSRC, "Guiding Principles of Reactor Operations."
5. WSRC, "Savannah River Site Reactor Operator's Code of Ethics."
6. WSRC, Inter-Office Memorandum (from J. C. Papalski to R. F. Anderson, et. al.), Subject: "Performance Indicator Report - April 1990 (U)," dated April 30, 1990.
7. WSRC, DOC #900334, "Reactor Maintenance Self-Assessment Report (U)."
8. WSRC, OPS-RMK#900580, Rev. 0, "Reactor Maintenance Improvement Plan Report (U)," dated April 27, 1990.
9. WSRC letter to F. McCoy III, "Power Operations in the 100's Areas," July 30, 1990.
10. WSRC, Inter-Office Memorandum (from E. P. Rahe, Jr. to R. F. Anderson, et. al.), Subject: "Reactor Operation's Mission and Guiding Principles," dated September 13, 1989.
11. WSRC, Inter-Office Memorandum (from M. Hyder to E. Rahe and C. Baker), Subject: "'Pats on the Back' Weekly Report," dated May 11, 1990.
12. WSRC, Procedure RDP 9.01, Rev. 1, "Management Monitoring," effective date August 28, 1990.

13. WSRC, K-Area Standing Instruction (KSI) No. D-50, Rev. 0, "K-Area Management Inspections," dated February 13, 1990.
14. WSRC, "Material Condition - 100-K Management Walkdown Program," dated March 27, 1990.
15. WSRC, ROD Housekeeping/Material Condition Control Procedure RDP 3.02.
16. WSRC, "Management Assessment Plan," dated December 14, 1989.
17. WSRC, Management Policy (MP) 1.11, "Open Communication of Employee Concerns," Rev. 0, April 1, 1989.
18. WSRC, Management Requirements and Procedures (MRP) 1.06, "Open Communications of Employee Concerns," Rev. 1, October 23, 1989.
19. WSRC, "Employee Concerns Program Action Plan," March 30, 1990.

2.3 Issue Management and Resolution

2.3.1 Background

In recent years, the reactors at the Savannah River Site (SRS) have undergone careful scrutiny in the area of reactor safety. Three major appraisals, which focused primarily on Three Mile Island (TMI) related issues were the Crawford Reports, the DuPont Post-TMI Assessment, and the Committee on Nuclear Safety Report. In addition, documented reviews by the Advisory Committee on Nuclear Facility Safety (ACNFS), the National Academy of Science (NAS), the Government Accounting Office (GAO), and DOE addressed issues related to the SRS reactors that have arisen in light of Chernobyl. In all, over 75 documents contain the results of assessments addressing issues that could affect the operation and safety of the SRS reactors.

The need to manage issues concerning the operation and safety of the SRS reactors was defined in the K-Reactor Restart Strategy (also known as the "Charlotte Criteria"). In addition, issues that were identified as a result of ongoing safety reviews, routine work initiators, and the results of Charlotte Criteria MS-15 work backlog required evaluation and resolution. Approximately 4000 potential issues were identified that involved SRS reactor deficiencies or recommendations. These issues were assigned to Westinghouse Savannah River Company (WSRC) for disposition, with oversight provided by the Savannah River Restart Special Projects Office (SRSPPO) of DOE.

WSRC established the Restart Issue Management Program (RIMP) to evaluate, disposition, and resolve nuclear safety issues prior to restart. Issue management under the RIMP is based on the reactor restart scope of work and on information provided by a schedule/cost assessment committee and a restart change control board. The RIMP is controlled by the operating contractor's Reactor Restart Division Administrative Procedure RDP 10.1, "Restart Issues Management" (Reference 1), which charters an Issues Management Committee (IMC) of experienced management-level personnel to disposition identified issues.

The purpose of the RIMP was to provide a continuous, complete, documented process for controlling the restart of K-Reactor and to segregate issues required for restart from issues that are part of a longer term program. Restart items from the RIMP were incorporated into the Reactor Operations Management Plan (ROMP); the longer term safety concerns were included in the Reactor Safety Improvement Program (RSIP). (See Section 2.4 of this report for a discussion of the RSIP.) As developing issues were identified, they were incorporated into the RIMP process and dispositioned accordingly.

2.3.2 Restart Criteria

The restart criteria for issue management establish the minimum requirements which must be satisfied to ensure that significant issues related to reactor operation are documented, brought to management attention, and corrected in an efficient manner. The criteria for determining the adequacy of the SRS issue management program for restart are described below.

(1) General

- (a) The contractor shall have in place and initiated an integrated issues management system to comprehensively identify, report, evaluate, track, trend, and disposition conditions or issues which have been determined to be potentially adverse to reactor/personnel safety. The system may be divided into elements or components provided that the methodology by which deficiencies, conditions, or issues are identified and relegated to a specific system component is accomplished consistently and in accordance with a formalized procedural control process for ensuring proper interaction with other system components.
- (b) Issues/deficiencies involving the material condition of the reactor plant and its supporting facilities, radiological protection, environmental impacts, and other conditions affecting safe reactor operation shall be included in the issues management system. Corrective actions shall be periodically and independently reviewed to ensure their effectiveness.
- (c) Where continued reactor operation is maintained concurrently with dispositioning of issues, appropriate compensatory measures shall be proposed for DOE concurrence on a schedule consistent with the safety significance of the issue.

(2) Issue Identification

- (a) The integrated issues management system shall contain mechanisms for the identification and reporting of conditions which could affect the safe operation of the reactor. Provisions shall also be in place to ensure that the individuals and supervisors are aware of the requirements for which they are responsible (i.e., the system shall clearly identify who is responsible for dispositioning the issue being tracked in the system).
- (b) Items to be considered by the system shall include (1) defective material, (2) defective or malfunctioning equipment, (3) personnel-caused errors, (4) administrative/training/procedural deficiencies, (5) operation outside established Technical Specification limits or Federal, State, or local government established limits (e.g., discharge permits), (6) new issues raised as a result of reactor operating events, and (7) other items deemed by the contractor to have an adverse effect on safe reactor operation.
- (c) Activities which shall be implemented to identify potentially adverse safety conditions should include, but not necessarily be limited to, periodic management plant walkdown inspections, internal audits and surveillances, and independent internal and external operating experience reviews.

(3) Issue Evaluation

The integrated issues management system shall have a mechanism similar to that required in the commercial nuclear industry under Title 10 of the Code of Federal Regulations (CFR) Part 50.59 to define the significance of an issue or condition to personnel/reactor safety. The system shall also provide a mechanism to determine the root cause of the condition, the remedial corrective action(s) or compensatory measure(s) to be taken, and action(s) to prevent recurrence. The results of such evaluations shall be provided to senior contractor management for review and transmission to the DOE.

(4) Staffing

- (a) The integrated issues management system shall be staffed with personnel whose expertise and training are commensurate with assigned duties and responsibilities. Responsibility for directing and ensuring the effective implementation of the program shall be at a senior level in the contractor's organization, having sufficient authority and accessibility to executive management.
- (b) Responsibility for the conduct of system operations shall be with organization managers associated with the function at issue. The contractor's departmental managers shall be responsible for ensuring the completeness, integration, and effectiveness of the portion of the program under their cognizance, including interfaces with other organizational departments.
- (c) Staffing required to initiate the program shall be established by restart. Full staffing may occur post-restart.

(5) Documentation

- (a) WSRC shall maintain documentation on each issue that shall include, as a minimum, the following: (1) identification of the issue, (2) the date the issue was identified, (3) the disposition of the issue, (4) identification of root causes, (5) corrective actions taken or required, (6) implementation dates for the corrective actions, and (7) the contractor staff member responsible for follow-up of the issue.
- (b) Provisions shall be made to provide DOE staff, as required, direct access to issues management documentation described above.

(6) Issue Processing

- (a) The integrated issues management system shall be controlled by administrative procedures that should appropriately describe the methodology and requirements for identifying, reporting, evaluating, dispositioning, tracking, and trending deficient conditions. The administrative procedures should define the threshold and scope of control necessary for the identification,

tracking, and disposition of issues with interfaces and corrective action responsibilities clearly delineated.

- (b) Issues processing shall contain a mechanism which delineates specifically how corrective actions are performed and controlled, including any interface or interaction with the RSIP and the plant operating review committee or its equivalent.
- (c) The issues management system shall have a process for prioritizing the dispositioning as well as the resolution of issues. Any condition which poses a potential threat to the safety of the reactors or significantly hampers site personnel in the performance of duties necessary for the safe operation of the reactors shall be dispositioned as soon as possible.
- (d) The appropriate DOE staff shall be notified if a significant concern is confirmed. (Some potential conditions may warrant notification before the problem is confirmed.)
- (e) The experience level requirements of personnel involved in the system shall be defined in the system's administrative procedures.

2.3.3 Evaluation

Issue Identification

The scope of issues evaluated by the operating contractor's Restart Issues Management Program (RIMP) includes safety issues identified by both internal and external audits, appraisals, and assessments. The sources include the Integrated Safety Evaluation Program (ISEP), the Nuclear Utilities Services (NUS) Operating History Review, the NAS Report, the DOE Environment, Safety and Health (ES&H) Technical Safety Appraisals, the Crawford Committee Reports, the Hamric Reports, ACNFS findings, the DOE Independent Review Committee Report, and the Westinghouse Independent Safety Recommendations (WISR). Also evaluated under the RIMP are potential safety issues identified by normal work processes, such as Reactor Technology Memoranda, SRL Green Letters, Requests for Engineering Assistance, Routine Work Initiators (for example, nonconformance reports, Reactor Tech memos, Request for Engineering Assistance, ES&H and QA findings, Modifications, Reactor Event Reports, and Unusual Occurrence Reports), and Facility Equipment Instructions.

The safety topics developed by the ISEP provide the breadth of issues chosen to assure completeness of the RIMP scope. The ISEP was initiated in August 1987 to review the design and operation of the SRS reactors, assessing and documenting the degree of conformance to current NRC and DOE criteria. The ISEP was based on the Systematic Evaluation Program (SEP) methodology used in the commercial nuclear industry to assess the safety of early vintage nuclear power plants in comparison to modern NRC requirements.

The first phase of the ISEP was the development of a list of safety topics applicable to SRS reactors. The ISEP candidate topic list contained a total of 2016 candidate safety-related topics including the 137 NRC Systematic

Evaluation Program topics. Additional topics were derived from TMI Action Items, NRC Generic Letters and Bulletins, SRS operating history review, ACRS letters, NRC Unresolved Safety Issues, and NRC and DOE reviews of the Chernobyl accident. The topic list was condensed to 184 topics by deleting duplicate and nonapplicable topics. The topics systematically developed by the ISEP represent the complete set of safety issues identified in the commercial industry that are applicable to SRS, as well as additional SRS-specific issues.

WISR recommendations were also included in the RIMP scope, addressing the capacity for continued safe reactor operation, required remedial actions, and post-restart corrective actions. A Westinghouse review committee, independent of Westinghouse Savannah River Company, concluded that safe restart could occur upon completion of a list of recommended upgrades to systems. WSRC is implementing the 24 WISR recommendations designated as pre-restart by the WISR committee. The remaining WISR recommendations were considered in the RIMP process.

Issue Evaluation

The RIMP process provides assurance that the SRS reactors can be operated without undue risk to the health and safety of the public. The RIMP criteria and associated evaluation process serve a role analogous to the NRC design/licensing process, but are limited in scope to the prevention and mitigation of significant risk contributors.

The RIMP process provides added safety assurance through the evaluation of the criteria and their application by a multi-discipline senior committee, the Issue Management Committee. The evaluation criteria are designed to assure implementation of the same fundamental design safety concepts considered important in the commercial sector: safe shutdown and core cooling, recognition of operator importance and limitations, maintenance of the operating envelope, and acceptable environmental impact. The judgement used in applying the evaluation criteria is based on standards of acceptable performance derived directly from the commercial experience of the issue evaluators and the Issue Management Committee. The RIMP process provides a realistic approach to the application of defense-in-depth philosophy to existing plants. The process is not constrained by arbitrary deterministic design standards and NRC Regulatory positions, nor is it constrained by narrow engineering approaches. Rather, it provides a framework for risk-based assessment, and the application of good engineering judgement to provide a documented basis for restart safety and future safety improvements.

Restart evaluations against the RIMP criteria consider the ability of human performance (staffing and training) and management systems (procedures, instructions, checklists, and administrative programs) to support the conclusions reached. If any one of the RIMP criterion is met, the issue is considered to entail a critical deficiency that must be addressed prior to restart. The Issue Management Committee may defer corrective actions to the post-restart period based on their considerations, including quantitative risk assessment results.

The following are the five restart criteria as used by the Issue Management Committee:

- (1) Does the issue prevent reactor shutdown, maintenance of required shutdown margin, long term cooling, or essential monitoring?
- (2) Does the issue require operator action in less than 10 minutes to prevent or mitigate the consequences of events described in the SRS safety analyses?
- (3) Does the issue cause operation outside the Technical Specifications?
- (4) Does the issue result in a reduction in the margin of safety as described in the SRS safety analyses?
- (5) Does the issue result in an impact on the environment which, in turn, would result in noncompliance with the requirements of current regulations or permits as a consequence of reactor operations?

The first criterion assesses the ability of essential safety features required for reactor shutdown and cooling to perform their intended function. This criterion is derived from the fundamental safety concepts of accident prevention and protection and represents the central element of the safety philosophy. Consideration of this criterion is not limited to the safety margins defined in the existing SAR and technical specifications; commercial standards and PRA insight are used in evaluating the reliability of the essential safety functions. Inclusion of minimum shutdown margin and post-trip monitoring in this criterion provides a completeness check that reactor shutdown and cooling can be maintained.

The second criterion assesses the ability of the plant operators to respond to plant conditions. This criterion is based on commercial experience in which operator action outside the control room is an important consideration to assuring satisfactory plant response to certain abnormal occurrences. Evaluation of this criterion considers the timing and complexity of operator actions to assure the burden on the operating staff is acceptable.

The third criterion evaluates whether the plant can be operated within the operating envelope defined by the technical specifications. Both this criterion and the fourth criterion call for a deterministic assessment of reactor safety by comparing potential plant conditions with existing safety analyses. The technical specifications provide the safety limits and limiting operational configurations. Challenging safety limits as a result of an identified issue would normally require corrective or compensatory action prior to restart.

The fourth criterion assesses the ability of the full complement of plant systems to perform their safety functions. By application of this criterion, the plant is shown to be within the bounds of the existing safety envelope for SRS reactors. Further, this process aids in identifying specific areas for improving the documentation of the SRS safety analysis.

The fifth criterion provides assurance that the environmental impact associated with the safety topic is within current environmental regulations. This criterion reflects increased sensitivity to providing assurance that environmental impacts are identified, evaluated, and managed. A checklist that considers the impact on air, surface water, ground water, and site use permits, is used to determine whether detailed environmental assessment is required in order to make final disposition on the environmental impact.

Staffing

The makeup of the Issue Management Committee represents functional areas within the WSRC Reactor Restart Division (RRD). There are a sufficient number of individuals to provide a diversified review of all items without involving anyone that might have a conflict of interest when dealing with a specific issue. The members are all seasoned, management-level personnel who, together, have significant experience with SRS and commercial reactors (an average of greater than 10 years SRS experience and greater than 20 years nuclear experience). The following is a listing of the titles and disciplines of the members:

<u>Discipline</u>	<u>Titles</u>
Technical Directorate	o Technical Director, Chairman o Manager of Nuclear Safety, Alternate Chairman
Strategic Planning	o Manager, Strategic Program o Manager, Strategic Projects
Operations, Training	o Operations Staff Assistant o Station and Issues and Improvements Program Manager
Engineering	o Manager of Technical Support Reactor Engineering o Manager of Reactor Engineering Support
Maintenance	o Central Maintenance Manager o Central Maintenance Assistant Manager o Engineering Service Manager
Assessments	o Manager of Operations Assessments o Process Engineer, Events Assessment
SRL Safety	o Manager of Reactor Safety Research, Alternate Chairman

- o Manager of Reactor Risk Assessments
- o Manager, Reactor Safety Control Reactor Risk Assessment

As exemplified by the performance of the Issues Management Committee, these individuals are of the highest caliber and have performed their function within the bounds of the issues management process with superior professionalism. Due to the involvement of these individuals the RIMP process has proven to be a strength of the SRS Restart Program.

Summary of RIMP Process Assessments

Various evaluations, assessments, and surveillances have been performed to monitor the RIMP process and assure that it has performed its intended function. During September 1989, DOE-SRSPO management performed a review of the RIMP process to assure that no issues of consequence were bypassed and that those issues identified received an appropriate and timely review. At the conclusion of this review, DOE-SRSPO was satisfied with the process and items reviewed.

During November 1989, WSRC performed an internal surveillance (by individuals internal to WSRC RRD) of the RIMP process. The surveillance was performed to ascertain the degree of procedural compliance within the RIMP process. During the comparison between the work being performed and the procedural requirements only minor deviations from procedures were noted. Those areas noted as deviations were quickly corrected and did not detract from the technical merit of the RIMP process.

During January and February 1990, an external audit by WSRC ES&H and QA division was conducted on the RIMP process. The audit was performed to evaluate the adequacy and effectiveness of the implementation of organizational management systems, hardware, and operational improvements identified as being required to safely restart and operate the SRS Reactors. A major portion of the audit dealt with verifying that RIMP items were being met via the RIMP process. The conclusion of the audit team was that the RIMP process had developed effective methods and procedures for collecting existing safety issues, performing and documenting in-depth technical reviews, and categorizing issues to ensure appropriate resolution was specified and effected.

During the first quarter of 1990, SCIENTECH performed an audit of the RIMP process. The purpose of this audit was to independently review a sampling of issues and determine if they had been adequately addressed. SCIENTECH concluded that the issues had been completely and adequately resolved.

Concurrent with the RIMP process, DOE-SRSPO has been reviewing approximately 30 percent of the RIMP packages to determine if all areas within the reviewed issues have been adequately addressed. The RIMP evaluations and actions have been, for the most part, concurred with by DOE-SRSPO. In those few areas where disagreements have developed, the RIMP process has readdressed the issue to resolve DOE-SRSPO concerns. The conclusion derived from these reviews is

that the RIMP process has adequately addressed the issues and has initiated appropriate corrective actions to resolve them.

An independent review of the WSRC RIMP was initiated by the DOE-SRSPO to ensure that issues identified by independent external and internal reviews were captured within the RIMP process and were properly dispositioned in accordance with the WSRC restart criteria. The review also included an evaluation of the disposition against the DOE SER restart acceptance criteria. The review ensured that:

- o Issues were being addressed and resolved in accordance with the SER Criteria.
- o The SER Criteria bounded the WSRC restart criteria.
- o Critical issues identified by the various external and internal review groups (WISR, NUS Operating History Review, the ES&H Technical Safety Appraisals, the Crawford Committee Reports, The Hamric Reports, the DOE Independent Review Committee Report, and the Savannah River Laboratory Green Letters) were properly addressed for restart.

The conclusions drawn from this review were as follows.

- o WSRC dispositioned items in accordance with their Restart Criteria.
- o The SER restart criteria bound the WSRC restart criteria. Of the 153 issues reviewed, only 7 issues were identified as being required for restart that did not have specific SER restart acceptance criteria. These seven issues generally discussed very specific component level issues, such as testing of chemicals for contaminants prior to addition to a system, modification of check valves, etc.
- o The Issues Management Committee's packages reviewed were complete in most cases and reflected a thorough and detailed analysis of the issue.

Documentation and Issues Processing

New issues, which were identified by internal SRS sources through surveillances and plant operation and by sources external to SRS, such as audit and inspection teams, continue to be addressed by the RIMP process. Eventually, the RRD Action Item Management System (AIMS) will replace RIMP, and a linking document is expected to formalize the transfer.

AIMS will programmatically identify all actions required of RRD, including reportability, evaluation, tracking, trending, disposition, and closure of the items when appropriate corrective action has been completed. The intent of this proposed program is to continue the management and control of issues as the restart issue management systems reaches completion. Both an Operational

Safety Review Committee (SER 9.1) and a Reactor Independent Safety Review Committee (SER 9.2) will use AIMS to programmatically identify all items with actions required of RRD. A charter for RRD AIMS has not yet been developed; therefore, the staff did not review such items as independent review, decision methodology, and compensatory measures for continued operation.

The staff has reviewed the AIMS Draft Implementation Plan, (Reference 2). In that plan AIMS categorizes issues for processing by the three component segments (i.e., external, surveillance, and plant operations). As reviewed, the draft document did not contain mechanisms for the identification and reporting of conditions that could affect the safe operation of the reactor. Responsibility for dispositioning of issues was not fully established, nor were the threshold and scope for elevating issues to a higher level review defined.

Notifications, reporting, determinations of corrective action, and unreviewed safety question (USQ) determination will be accomplished within the RISRC. A root cause determination was not addressed in the draft implementation plan.

WSRC has not yet submitted staffing and training plans. Therefore, the staff has not reviewed the staff's expertise or the proposed staff training to ensure that they are commensurate with assigned duties and responsibilities.

Issue documentation was not defined within the proposed implementation plan. Administrative procedures are required that will control the integrated management system and define specifically how issues are processed, how corrective actions are performed and controlled, and how interface and interaction is controlled. These procedures have not been fully developed.

The staff has reviewed the progress being made by WSRC's Reactor Restart Division (RRD) in accordance with the above stated restart criteria. On the basis of that review of draft WSRC procedures, the staff has determined that formation of an AIMS Working Group and the administrative procedure upgrades to support the integrated issues management system are key elements of issues management and resolution. The staff determined that an integrated issues management system is under development; however, RRD's efforts in this undertaking have not yet been developed to the point where a more complete evaluation can be performed relative to the DOE restart criteria. The completion of this review is an open item.

2.3.4 Conclusion

The RIMP process was initiated and is being conducted in a manner that assures the relevant SRS specific issues are being addressed. Through the rigorous application of comprehensive RIMP evaluation criteria, potential safety issues were dispositioned. The completeness of the RIMP process ensured that restart sensitive issues were identified in a timely manner. The overall success of this program is due, in part, to the contribution made by the Issue Management Committee. As supported by the various audits, assessments and surveillances, the RIMP process has proven to be very thorough.

The proposed RRD AIMS program, although not yet fully developed, should define a sound approach for the management of issues related to safe operation of the reactors. Finalization of the draft implementation plan for the AIMS program and related implementing instructions and procedures should address the findings identified above. Pending full development of the AIMS program, issue resolution remains an open item. The DOE staff will update its evaluation and document the results in an SER Supplement.

2.3.5 References

1. WSRC, RDP 10.1 Rev. 0, "Restart Issue Management," DPSP 87-1211.
2. WSRC, "Draft RRD Action Item Management System (AIMS) Implementation Plan," March 18, 1990.

2.4 Reactor Safety Improvement Program (RSIP)

2.4.1 Background

The Reactor Safety Improvement Program (RSIP) is the management program that will be used to implement reactor safety improvements after restart of the Savannah River Site (SRS) reactors. This program will be used to schedule ongoing improvements as well as those to be implemented during outages. The RSIP is the primary element in planning for the accomplishment of improvements and providing justifications for the budgetary requests required to implement these improvements. RSIP provides a comprehensive approach to improved reactor safety at SRS and will be a "living program" that will continue to develop as new problems are identified and solutions are developed. The objectives of the RSIP are as follows:

- o Improve the human performance (including training and procedures) and management systems at SRS so that they are as good as, or better than, the best in the commercial nuclear industry.
- o Improve plant hardware so that the risk to the general public from SRS reactor operations is no more than that of the older commercial pressurized water or boiling water reactors.

Meeting these objectives will be a continuing process involving efforts to meet DOE safety guidelines and requirements and minimizing the risk of reactor operations.

2.4.2 Restart Criteria

The criteria for determining the adequacy of the RSIP for restart are described below.

(1) General

- (a) WSRC shall implement a Reactor Safety Improvement Program (RSIP) to provide a decision-making process for focusing and tracking the scope, priority, schedule, and budget of programs and improvements made to the SRS reactors for enhancing reactor safety, environmental safety, safeguards, and security. The program shall address improvements to be made as a part of and subsequent to reactor restart, including those identified in the Reactor Issues Management Program (RIMP) as being necessary for long term improvement.
- (b) The RSIP is to be predicated on the "Integrated Living/Implementation Schedule" process used in the commercial nuclear industry and adapted as necessary to the unique needs of the SRS reactors.
- (c) The RSIP is to be the principal component for the determination and scheduling of improvements to be made during reactor outages. The prioritized improvements in the RSIP shall be balanced against

resource constraints and used to determine the specific improvements to be performed during the outages in a manner consistent with the guidance provided in Institute of Nuclear Power Operations report INPO-87-027, Guidelines for the Management of Planned Outages at Nuclear Power Stations.

(2) Definition of Improvements

- (a) The improvements described and managed in the RSIP shall include any hardware modifications, technical studies, management systems, and human performance initiatives defined as reactor safety-related, including those identified in RIMP.
- (b) The addition of new improvements shall be made by a formal process that will require work scope approval jointly by the WSRC/DOE. To assist the decision-making process for newly added improvements, the contractor's estimate of the minimum expenditure of resources shall be stated when new improvement items are added to the RSIP. Also to be included is WSRC's evaluation of the significance and expected results of performing improvements that are identified currently and those proposed for the future, which shall provide the technical justification for the improvement. This improvement evaluation/ranking process shall undergo several independent reviews with the final ranked listing of improvements predicated on a numerical value derived from the weighted attributes developed from all reviews.
- (c) The ranking of improvements shall be reviewed periodically by the contractor and re-rated for any changes in work scope, newly added improvements, etc., and the RSIP updated accordingly with the concurrence of the DOE.

(3) Scheduling

- (a) WSRC shall develop Level 2 schedules for improvements contained in the RSIP that will begin within a 3-year period after reactor restart. These schedules are to be used as the primary tool for developing outage schedules, capital resource requirements, annual budgets, and long range budget forecasts for reactor improvements.
- (b) The schedules developed for improvements are to be approved by (1) the responsible line manager, (2) the contractor's planning, scheduling and cost control manager, (3) the RSIP manager, and (4) the DOE. Scheduling constraints (e.g., manpower and budget resources, outage durations) are to be determined jointly by the contractor's RSIP organization, outage planning organization, and the DOE.

- (c) Administrative procedures shall be developed by the contractor to (1) control the scheduling of planned outages required for timely implementation of improvements and (2) describe the process to be used for the development of resource requirements (costs and manpower).

(4) RSIP Management

- (a) The RSIP is to be the contractor/DOE interface for reactor safety improvements and for providing project direction. The organization created to manage the RSIP shall have the responsibility of (1) maintaining an accurate technical/schedule status, (2) ensuring timely completion of improvements, (3) identifying problems to senior management, and (4) periodically keeping management abreast of program status through meetings and written reports. The RSIP organization shall also be responsible for the development of budgetary requirements for the successful implementation of improvements and the method for doing this shall be proceduralized.
- (b) A Reactor Commitment Tracking System shall be used to track all improvement items and their major schedule milestones. The RSIP organization shall be staffed with contractor personnel appropriately trained, experienced, and managed under the direction of a Project Manager.

2.4.3 Evaluation

General

WSRC published a document describing the RSIP (Reference 1), which was approved by WSRC's Deputy General Manager for Strategic Programs in July 1989; however, the RSIP has undergone many changes since that time. An updated document, the "Reactor Safety Improvement Program Summary Document" (Reference 2), which was issued the Strategic Programs Department, better describes the RSIP and provides detailed programmatic descriptions of the various aspects of the program. This document was approved by the Vice President and General Manager of the Reactor Restart Division (RRD) in October 1990.

The RSIP establishes a comprehensive process for focusing and tracking improvements to the SRS reactors. Newly identified improvements are evaluated initially by the WSRC Issue Management Committee (IMC). When the IMC defines an improvement item as a long-term program or project, the RSIP becomes the focal point for managing the issue from its inception through its closure.

Currently, the RSIP process is used for tasks which are to be completed as part of and subsequent to restart. All items identified as nonrestart items in the RIMP were placed into the RSIP in accordance with the RSIP section (RDP 11.01) of the WSRC RRD procedures manual (Reference 3). RSIP procedures define the processes used by the RSIP engineers in identifying the scope, schedule, and prioritization of an improvement. Section 4.0 of RDP 11.01 contains the specific procedures for accepting items into RSIP.

The RSIP is based on the "Integrated Living/Implementation Schedule" process used in the commercial industry. The RSIP program will continue to be used throughout the life of the SRS reactors. Similar to the "Integrated Living/Implementation Schedule" process, the RSIP permits newly identified items to be incorporated into the overall RSIP scope and provides for the accomplishment of individual tasks according to their priority. The "Integrated Living/Implementation Schedule" concept ensures resources are focused on the tasks which provide the maximum benefit rather than on uncoordinated expenditures by individual line managers.

Reactor outage schedules are developed by WSRC's Outage Support Group in coordination with the RSIP organization. In accordance with Procedure RSIP 4-5 of RDP 11.01, "Outage Schedules," RSIP develops schedules for the individual items and recommends which items should be worked during which particular outage. The RSIP organization's outage recommendations are provided to the WSRC Pre-outage Planning Group. Both organizations function in accordance with guidelines set forth in INPO-87-027.

The "RSIP Summary Document" contains a description of how the Strategic Programs Department conforms to INPO-87-027. Basically, INPO-87-027 recommends that outage details be planned at least 1 year in advance and that several future outages be maintained in the developmental stages while the upcoming outage is being planned. The Strategic Programs Department conforms to these recommendations in that detailed scopes, schedules, and tasks are developed for work to be performed in the next 3 years, and less-detailed information is developed for work to be performed outside of that 3-year period.

Definition of Improvements

WSRC has upgraded the procedures for incorporating new improvements into the RSIP process. These procedures are fully comprehensive and allow RSIP to cover a large spectrum of improvements that may involve hardware modifications, technical studies, management systems, and human performance initiatives, including those previously identified in the RIMP. The numbering system used in the RSIP can be used to distinguish between the various types of improvements. Each item is identified by a letter corresponding to its work type (e.g., E=Engineering Study, O=Operations Upgrade, T=Training or Procedure Upgrade, etc.).

WSRC uses the RSIP to manage and track improvement items that are safety related, as well as nonsafety related. For each proposed improvement the Strategic Programs Department creates a Project Control Plan (PCP) that summarizes the project scope, priority, budget, schedule, and manpower requirements. The PCP contains the contractor's evaluation of the minimum expenditure of resources and the overall significance of the project. The PCP provides a detailed compilation of information relative to the improvements that technically justifies the improvements.

The "RSIP Summary Document" outlines the method by which the PCP rating of each project is approved by the WSRC Management Review Committee and submitted for approval by the DOE. A June 1990 inspection by the DOE staff found that WSRC's RSIP procedures did not adequately cover the processes to be used to

obtain DOE approval of RSIP plans and schedules. Since the time of this inspection, the procedures have been revised, and obtaining DOE approval is acceptably addressed.

All RSIP items are ranked relative to overall benefit, and the rankings are used in the scheduling process. The total list of ranked RSIP improvements will be re-evaluated and re-ranked periodically. These re-evaluations will undergo several independent reviews to ensure a high level of acceptability for the completion of identified improvements.

In the spring of 1991, the Strategic Programs Department will begin issuing the "RSIP Master Plan and Schedule Report." In accordance with RSIP procedures, this Master Plan will be reissued each spring and will be used as a tool in the budget request effort. The "RSIP Master Plan and Schedule Report" will be approved by WSRC upper management and by the DOE.

The ranking of RSIP improvements is accomplished through the use of a benefit rating system that uses weighting factors. These weighting factors are comprehensive in nature and consider key factors that are important to the contractor/DOE decision making process. The current relative weight of each of these areas is provided below along with a list of the attributes:

- Reactor/Public Safety (29 percent)
- Environment (13 percent)
- Oversight and External Relations (13 percent)
- Industrial Safety (12 percent)
- Production (12 percent)
- Safeguards and Security (9 percent)
- Personnel Efficiency (7 percent)
- Cost (5 percent)

These weights were determined based on the consolidation of the results of a survey of WSRC and DOE management judgement as to the relative importance of each item. Since the RSIP is a living program, the relative importance of the attributes of the weighting system may be revised as a result of changes in priorities or programmatic goals. The results of this survey are documented and maintained by the RSIP engineer responsible for the computerized benefit rating system.

Scheduling

The WSRC Strategic Programs staff has developed PCPs for RSIP items which are scheduled to begin in FY91, FY92, and FY93. These PCPs contain information such as background, justification, objectives, Level 2 schedules, and associated resource requirements for each item. The RSIP staff is currently working to complete PCPs for items scheduled to begin in FY94. The FY94 PCPs will be completed by March 31, 1991, which will meet the restart criteria requirement of having scopes and Level 2 schedules developed for items that are to be worked within 3 years after restart.

The Capital Assets Management Plan (CAMP) is the SRS improvement plan encompassing planned work for the site over the next 5 years. The CAMP is considered a subset of RSIP. Because the amount of information developed for CAMP items is less than that required for the RISIP, reactor improvement items identified in the CAMP are fully documented and developed by the RSIP.

RDP 11.01 outlines the method by which WSRC uses the PCP information to develop outage schedules, capital resource requirements, annual budgets, and long range budgets. RSIP provides the baseline information used in the development of these requirements.

Procedures have been developed to ensure that scopes and schedules for RSIP improvements are approved by the responsible line manager (i.e., the manager of the Department of Planning, Scheduling, and Cost Control; and the manager of RSIP, the Strategic Programs Manager). Final schedule approval is by the Deputy General Manager for Strategic Programs and the DOE. This approval chain is outlined in Attachment F of RDP 11.01. Essentially, this attachment requires that PCPs be routed through the appropriate managers once they have been developed by Strategic Programs Department engineers in conjunction with line engineers. After the necessary WSRC managers have signed them, a Management Review Committee will approve the PCPs before they are submitted to the DOE. DOE's approval of the PCPs will occur via approval of the Master Plan and Schedule Report, which will be issued annually during the spring.

Schedules are constrained by projected manpower and budget resource availability along with duration of outages, as defined by interaction of the RSIP organization and DOE. This ensures that the higher priority items are accomplished within realistic budgetary constraints. RSIP procedures were revised in June 1990 to reflect how this information is used in the budget process.

RSIP Management

RSIP acts as the focal point for contractor/DOE interface and for providing project direction. Monthly meetings between DOE and the RSIP organization and meetings between DOE and the RSIP staff and outage planning staff, serve to maintain management cognizance over future improvement programs and projects, ensuring that the most desirable safety standards are maintained.

The WSRC RSIP organization has the responsibility for maintaining an accurate technical/schedule status and ensuring timely completion of improvements (Reference 4). This responsibility is addressed in RDP 11.01. In addition, the RSIP organization, through interactions with the reactor line organization, is responsible for the development of budget requirements to successfully implement the RSIP. The processes used for budget development requirements are found in RDP 11.01.

In September 1990, the DOE staff performed an analysis of the RSIP staff's use of the Reactor Commitment Tracking System. The staff found that RIMP "Priority 3" items are the only items specifically deemed to be a "commitment." That is, the RIMP priority 3 items are those issues identified by the RIMP evaluation process as being post-restart items with a specific

time-schedule commitment. There is a separate tracking system managed by the RIMP organization to track these "Priority 3" items. However, these items are also tracked, along with all other RSIP items, by the WSRC RSIP staff. On the basis of annual updates, monthly reports, and the Commitment Tracking and Improvement Prioritization System (CTIPS) database (which maintains the up-to-date status of all RSIP items), the RSIP staff maintains full cognizance of the status of all RSIP improvement items.

To keep management abreast of program status, the RSIP organization issues monthly reports and holds a monthly management meeting to discuss the current status. In addition, the DOE staff performs monthly inspections to ensure the program remains effective, and conducts monthly exit meetings with WSRC's senior management to discuss the DOE staff's observations and concerns regarding the program.

The RSIP Project Managers have the experience and training appropriate to their assignment. Their experience includes previous performance in scheduling reactor programs with Westinghouse Corporate and various other organizations. A June 1990 inspection by the DOE staff revealed that the Strategic Programs Department had no formal training program set up to educate the department's engineers about the various operational aspects of the SRS reactors. Since then, however, the Strategic Programs Department has established and adequately implemented such a training program. This program is designed to improve the performance of personnel on the basis of individual needs. A matrix of available courses outlines all required and recommended training and is used to provide the status for each person's completed training. Among the list of courses are reactor systems courses, general employee training, radiological worker training, and various other courses related to operational aspects of SRS.

The RSIP organization is currently being staffed to adequately handle most phases of the RSIP. The RSIP organization has been approved for the necessary personnel to allow the organization to function effectively. As the RSIP staff works to fill the open positions, they are supported by subcontractors to ensure the completion of necessary work.

The Strategic Programs Department has developed a formalized training program to enhance the effectiveness of their personnel. Training records are kept up-to-date to reflect the current status of each individual's requirements.

2.4.4 Conclusion

The DOE staff has determined that WSRC fully complies with the restart criteria for RSIP.

2.4.5 References

1. WSRC-RP-89-379, "Development and Management of Reactor Safety Improvement Program," July 1989.
2. WSRC-RP-90-99 Rev. 1, "Reactor Safety Improvement Program Summary Document," October 1, 1990.

3. WSRC RD-1, "WSRC Reactor Restart Division Procedures Manual"
4. WSRC-RP-89-384, "Reactor Safety Improvement Program Scopes and Schedules For Top 50 Ranked Improvements," July 1989.

2.5 Contractor Compliance with DOE Orders

2.5.1 Background

Historically, DOE has communicated its performance requirements to operating contractors through its operating contracts. The previous contractor for SRS was required to comply with safety-related DOE Orders under its operating contract; however, strict compliance to these contractual obligations was not enforced. When WSRC assumed its role as operating contractor on April 1, 1989, the operating agreement contained contractual requirements to comply with DOE Orders.

2.5.2 Restart Criteria

The criteria for determining the adequacy of the SRS program for contractor compliance with DOE Orders for restart are described below.

(1) Implementation and Assessment Program

The contractor shall have a program for implementing and assessing all contractor-applicable DOE Orders, with a senior manager designated as responsible for coordinating the site-wide activities necessary to ensure that compliance is achieved. This senior manager will interface with a designated counterpart, coordinate review of the Orders, and be responsible for ensuring that programs are established and implemented to achieve compliance. Accordingly, line management responsibilities for implementing policies, programs, and procedures to comply with DOE Orders shall be clearly defined.

(2) Demonstration of Compliance

The contractor shall comply with all DOE Orders applicable to reactor restart as determined by the Order Compliance Joint Task Force (OCJTF) to be Category I Orders (Appendix I). Documented evidence of compliance shall be developed and be available for review or Compliance Schedule Approvals (CSAs) or exemptions shall be provided to and approved by DOE. If compliance cannot be fully attained before reactor restart, CSA or exemption requests shall be submitted to and approved by DOE. Compliance includes the development and implementation of policies, programs, and procedures used to ensure compliance with DOE Orders as well as implementation and verification of compliance. Requests for CSAs shall follow the procedure in the January 18, 1990, memorandum (Reference 1) from SRRO to SRSP0 for the preparation and processing of CSAs.

2.5.3 Evaluation

Implementation and Assessment Programs

The staff reviewed the WSRC implementation and assessment programs. The purpose of these programs is to implement DOE Orders as they are issued and to

assess compliance with Orders issued prior to April 1, 1989. The results of the staff's review are discussed below.

As stated in the WSRC contract (Reference 2), a Contract Administration Notice (CAN) is the official mechanism for transmittal of Orders that DOE proposes for implementation by Savannah River Site (SRS) contractors. Upon receipt of a CAN, WSRC must promptly review the transmitted regulation. If the requirement can be accomplished within existing funding, WSRC shall immediately implement the regulation or submit an implementation plan to DOE. WSRC has committed to assess compliance with all DOE Orders issued prior to April 1, 1989, regardless of the issuance of a CAN by DOE.

On January 1, 1991, all SR Orders were canceled (Reference 3). Directive Implementation Instructions (DIIs) will be issued when further formal, administrative information regarding DOE Directives is necessary. When a DOE Order is issued, DOE will transmit the order via a CAN as soon as it is received onsite.

WSRC uses the procedure MRP 3.02, "DOE Directives Administration" (Reference 4), as the guideline for reviewing and implementing DOE Orders. Issuance of the CAN initiates the compliance requirement; however, MRP 3.02 does not discuss the use of the CAN system in WSRC's administration of the DOE Orders. MRP 3.02 does not provide any guidance to staff managers from whom this procedure requires action to establish and document criteria for meeting Order requirements. Guidance provided in MRP 3.33, "Performing DOE Directive Compliance Self-Assessments and Generating Associated Documentation Submittals" (Reference 5), should be referenced in MRP 3.02 when MRP 3.33 is finalized.

During the transition period between contractors, WSRC performed an initial assessment of compliance with those DOE Orders issued prior to April 1, 1989. This assessment resulted in the transmittal of a report of WSRC's findings to the DOE staff on March 31, 1989 (Reference 6). On September 29, 1989, WSRC issued the "ESH&QA-Related Order Compliance Plan" (Reference 7), which specifically addressed compliance with DOE environment, safety, health and quality assurance (ESH&QA) Orders.

The ESH&QA-Related Order Compliance Plan detailed the status of compliance with specific requirements of all DOE and DOE-SR Orders and Notices related to ESH&QA that were applicable to WSRC and were issued prior to October 1, 1989. The plan is mainly WSRC's management assessment of its own management systems (i.e., policies, higher-tier procedures, and programs); actual implementation was not discussed. WSRC has stated that it recognizes the deficiencies in this plan and that it will revise the plan to resolve these deficiencies as well as to delete all references to DOE-SR Order requirements. The submittal date for a revised plan was extended to December 31, 1990; however, only a few final self-assessments have been submitted for staff review. To verify compliance with DOE Orders, WSRC must complete the review of its policies, programs, and procedures and their implementation, and document the results in a final controlled self-assessment. This remains as an open item.

Orders that WSRC identified as applicable to WSRC, but not ESH&QA-related, were assessed through a parallel effort that resulted in the issuance of "DOE Directives Baseline Management Systems Assessment Phase 2 Final Report" (Reference 8). This report, which details WSRC's assessment of non-ESH&QA-related Orders, is similar to the ESH&QA-Related Order Compliance Plan in structure. The staff determined that the lists of DOE Orders identified in both of these WSRC assessments are incomplete. This is an open item.

During discussions with WSRC, the staff determined that most of the differences between the staff's list of Orders applicable to WSRC and WSRC's list of applicable Orders resulted from WSRC's decision to exclude the following:

- (1) Orders that did not address specific contractor requirements, but rather set forth requirements as to how DOE was to do business.
- (2) Orders that required specific actions on the part of the Field Office with which the Field Office could not be in compliance unless certain actions were performed by the contractor.
- (3) Orders that did not state specific contractor requirements and did not provide more than general guidance on DOE practices.

The staff has stated that the contractor must comply with the list of DOE Orders related to safety, safeguards and security referred to as Level I DOE Orders (Appendix II). The WSRC revised self-assessment has not been finalized, and there has been no other mechanism by which WSRC has stated that it would assess its compliance against all of the Level I DOE Orders.

Demonstration of Compliance

DOE established the Order Compliance Joint Task Force (OCJTF) on November 7, 1989, to verify WSRC's demonstration of Order compliance. The activities of the OCJTF are described in the OCJTF Charter (Reference 9). The OCJTF is composed of representatives from DOE Headquarters, SRSP0, and SR. The OCJTF categorized the DOE Orders for the Savannah River Site as follows:

Category I -	<u>Pre-reactor-restart</u> -- This category will include Order requirements applicable to the first cycle of reactor operations and those aspects of site operations directly applicable to the first cycle of reactor operations.
Category II -	<u>Post-reactor-restart</u> -- This category will include remaining Order requirements applicable to reactor operations and those aspects of site operations directly applicable to reactor operations.
Category III -	<u>Environment, Safety & Health, Quality Assurance and Safeguards & Security (S&S)</u> -- This category will include remaining ESH&QA and S&S Order requirements on

a site-wide basis, beyond those requirements specifically related to reactor operations.

Category IV - Administrative and Miscellaneous -- This site-wide category includes remaining Order requirements not previously addressed in compliance review and verification efforts related to reactor operations, ESH, QA and S&S.

Category I Orders were determined by screening DOE Orders against the criteria shown in Figure 2.5.1 in an attempt to prioritize the Orders for scheduling the compliance reviews.

As stated in Section 2.5.3.1, the staff has reviewed the proposed WSRC schedules for verification of compliance with Category I Orders and has determined that they do not fully meet the restart criteria. In several areas, WSRC's verifications are intended to be completed as part of the normal periodic QA surveillances. This results in verifications not being completed in a timely manner. WSRC should submit its plans for verification of compliance with DOE Orders to the staff for approval. This remains as an open item.

The OCJTF has performed independent DOE verifications of Category I Orders per the "Management Plan - DOE Order Compliance Verification Programs at the Savannah River Site" (Reference 10). WSRC shall develop responses for the noncompliant items identified in the Verification Reports issued at the conclusion of each Order verification. Responses are in the form of a request for a Compliance Schedule Approval (CSA), an exemption or a short-term compliance statement (STCS). All CSAs and exemptions shall be approved or corrective actions approved and completed for all Category I Order noncompliant items prior to K-reactor restart. This remains as an open item.

MRP 3.33, "Performing DOE Directive Compliance Self-Assessments and Generating Associated Documentation Submittals," was drafted for the development and submittal of WSRC self-assessments and requests for CSAs, exemptions and STCSs. This procedure has never received management signature and does not yet adequately address the items to be included in the WSRC requests, as required by the DOE procedure. WSRC is currently incorporating DOE comments into its procedure to more accurately reflect the content that DOE requires to be contained in the respective documents. This remains as an open item.

2.5.4 Conclusion

Based on the above evaluation, the DOE staff concludes that WSRC has the framework for an Order compliance program but that it has difficulty in implementing the program's requirements due to a lack of senior management commitment for the process. Furthermore, current WSRC policies do not consolidate the Order compliance program in a manner that is readily understandable and recognizable as an ongoing process. The WSRC plans for verification of compliance have not been provided to DOE. In addition, there is no approved schedule for WSRC submittal of requests for CSAs or exemptions or completion of corrective actions. This does not allow sufficient time to

process CSAs or exemptions or to take alternate action for Category I Order noncompliant items prior to K-reactor restart. In addition, WSRC must commit to performing self-assessments against all of the Level I DOE Orders. WSRC must complete and submit a final controlled self-assessment for all Category I DOE Orders prior to K-reactor restart.

2.5.5 References

1. "Procedure for Contractor Requests for U.S. Department of Energy (DOE) Directive Compliance Schedule Approvals and Exemptions," January 18, 1990.
2. Contract No. DE-AC09-88SR18035 Between U.S. Department of Energy Savannah River Operations Office and Westinghouse Savannah River Company for Management and Operation of the Savannah River Plant, Effective Date: April 1, 1989.
3. U.S. Department of Energy Savannah River Operations Office Notice SR N 1321.1E, September 13, 1990.
4. Management Requirements and Procedures 3.02, Rev. 0, "DOE Directives Administration," April 1, 1989.
5. MRP 3.33, "Performing DOE Directive Compliance Self-Assessments & Generating Associated Documentation Submittals," Draft, Unsigned-interim.
6. Letter from L. S. Facchini (WSRC) to E. S. Chaput (DOE-SR), "Final Transition Report on the DOE Directives Assessment by WSRC Management," dated March 31, 1989.
7. "Westinghouse Savannah River Company ESH&QA-Related Order Compliance Plan," WSRC-RP-89-738, September 30, 1989.
8. "DOE Directives Baseline Management Systems Assessment Phase 2 Final Report (U)", WSRC-RP-89-969, September 29, 1989.
9. "Order Compliance Joint Task Force (OCJTF) Charter," Rev. 1, July 15, 1990.
10. "Management Plan - DOE Order Compliance Verification Program at the Savannah River Site," July 19, 1990 (approved by S. D. Richardson and E. J. McCallum).

CRITERIA FOR DETERMINATION OF CATEGORY I ORDERS

2.5-6
Figure 2.5-1

APPENDIX I

LEVEL I CATEGORY I DOE DIRECTIVES SAVANNAH RIVER SITE

LEVEL I CATEGORY I DOE ORDERS
Savannah River Site

<u>Order #</u>	<u>Subject</u>
1. 1360.2A	Unclassified Computer Security Program
2. 5000.3A	Occurrence Reporting and Processing of Operations Information
3. 5400.1	General Environmental Protection Program
4. 5400.3	Hazardous and Radioactive Mixed Waste Program
5. 5400.4	Comprehensive Environmental Response, Compensation, and Liability Act Requirements
6. 5440.1C	Implementation of National Environmental Policy Act
7. 5480.4	Environmental Protection, Safety and Health Protection Standards
8. 5480.5	Safety of Nuclear Facilities
9. 5480.6	Safety of DOE-Owned Reactors
10. 5480.7	Fire Protection
11. 5480.11	Radiation Protection for Occupational Workers
12. 5480.18	Training Accreditation
13. 5481.1B	Safety Analysis and Review
14. 5484.1	Environmental Protection, Safety and Health Protection Information Reporting Requirements
15. 5500.1A	Emergency Management System
16. 5500.2A	Emergency Notification, Reporting Requirements
17. 5500.3	Reactor and Non-Reactor Nuclear Facilities Emergency Management Planning, Preparedness and Response Program for Department of Energy Operations
18. 5500.4	Public Affairs Policy and Planning Requirements for Emergencies
19. 5630.11	Safeguards and Security Program
20. 5630.13	Master Safeguards and Security Agreements
21. 5631.2B	Personnel Security Program
22. 5631.6	Personnel Security Assurance Program
23. 5632.2A	Physical Protection of Special Nuclear Material and Vital Equipment
24. 5632.3B	Operations Security
25. 5632.7	Protective Forces
26. 5632.9	Issuance, Control and Use of Badges, Passes and Credentials
27. 5633.3	Control and Accountability of Nuclear Material
28. 5700.6B	Quality Assurance

APPENDIX II

LEVEL I DOE DIRECTIVES

LEVEL I DOE ORDERS

Order #	Subject
1. 1270.2A	Safeguards Agreements with the International Atomic Energy Agency
2. 1360.2A	Unclassified Computer Security Program
3. 1360.4A	Scientific and Technical Computer Software
4. 1540.2	Hazardous Material Packaging for Transport - Administrative Procedures
5. 1540.3	Base Technology for Radioactive Material Transportation Packaging Systems
6. 1540.4	Physical Protection of Unclassified, Irradiated Reactor Fuel In Transit
7. 3791.2A	Federal Employee Motor Vehicle Safety Program
8. 4330.XXX	Maintenance Management Program
9. 4700.1	Project Management System
10. 5000.3A	Occurrence Reporting and Processing of Operations Information
11. 5300.2C	Telecommunications: Emission Security (TEMPEST)
12. 5300.3B	Telecommunications: Communications Security
13. 5300.4B	Telecommunications: Protected Distribution Systems
14. 5400.1	General Environmental Protection Program
15. 5400.2A	Environmental Compliance Issue Coordination
16. 5400.3	Hazardous and Radioactive Mixed Waste Program
17. 5400.4	Comprehensive Environmental Response, Compensation, and Liability Act Requirements
18. 5400.5	Radiation Protection of the Public and the Environment
19. 5440.1C	Implementation of National Environmental Policy Act
20. 5480.1B	Environment, Safety and Health Program
21. 5480.3	Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances and Hazardous Wastes
22. 5480.4	Environmental Protection, Safety and Health Protection Standards
23. 5480.5	Safety of Nuclear Facilities
24. 5480.6	Safety of DOE-Owned Reactors
25. 5480.7	Fire Protection
26. 5480.8	Contractor Occupational Medical Program
27. 5480.9	Construction Safety and Health Program
28. 5480.10	Contractor Industrial Hygiene Program
29. 5480.11	Radiation Protection for Occupational Workers
30. 5480.13	Aviation Safety
31. 5480.15	Department of Energy Laboratory Accreditation Program for Personnel Dosimetry
32. 5480.16	Firearms Safety

LEVEL I DOE ORDERS

	<u>Order #</u>	<u>Subject</u>
33.	5480.17	Site Safety Representatives
34.	5480.18	Training Accreditation
35.	5480.19	Conduct of Operations Requirements for DOE Facilities
36.	5481.1B	Safety Analysis and Review
37.	5482.1B	Environment, Safety, and Health Appraisal Program
38.	5483.1A	Occupational Safety and Health Program for DOE Contractor Employees at Government-Owned Contractor-Operated Facilities
39.	5484.1	Environmental Protection, Safety and Health Protection Information Reporting Requirements
40.	5500.1A	Emergency Management System
41.	5500.2A	Emergency Notification, Reporting Requirements
42.	5500.3	Reactor and Non-Reactor Nuclear Facilities Emergency Management Planning, Preparedness and Response Program for Department of Energy Operations
43.	5500.4	Public Affairs Policy and Planning Requirements for Emergencies
44.	5500.7A	Vital Records Protection Program
45.	5530.1	Response to Accidents and Significant Incidents Involving Nuclear Weapons
46.	5600.1	Management of Department of Energy Weapon Program and Weapon Complex
47.	5610.1	Packaging and Transportation of Nuclear Explosives, Nuclear Components, and Special Assemblies
48.	5610.2	Control of Weapon Data
49.	5610.3	Program to Prevent Accidental or Unauthorized Nuclear Explosive Detonations
50.	5630.8	Safeguarding of Naval Nuclear Propulsion Information
51.	5630.11	Safeguards and Security Program
52.	5630.12	Safeguards and Security Inspection and Evaluation Program
53.	5630.13	Master Safeguards and Security Agreements
54.	5630.14	Safeguards and Security Program Planning
55.	5631.1A	Security Education Program
56.	5631.2B	Personnel Security Program
57.	5631.4	Control of Classified Visits
58.	5631.5	Violation of Laws, Losses, and Incidents of Security Concerns
59.	5631.6	Personnel Security Assurance Program
60.	5632.1A	Protection Program Operations

LEVEL I DOE ORDERS

	<u>Order #</u>	<u>Subject</u>
61.	5632.2A	Physical Protection of Special Nuclear Material and Vital Equipment
62.	5632.3B	Operations Security
63.	5632.5	Physical Protection of Classified Matter
64.	5632.6	Physical Protection of DOE Property and Unclassified Facilities
65.	5632.7	Protective Forces
66.	5632.8	Protection Program Operations - Systems Performance Tests
67.	5632.9	Issuance, Control and Use of Badges, Passes and Credentials
68.	5633.2	Control and Accountability of Nuclear Materials, Responsibilities and Authorities
69.	5633.3	Control and Accountability of Nuclear Material
70.	5633.4	Nuclear Transactions: Documentation and Reporting
71.	5633.5	Nuclear Materials Reporting and Data Submission Procedures
72.	5634.1A	Facility Approval, Security Surveys, and Nuclear Materials Surveys
73.	5635.1A	Control of Classified Documents and Information
74.	5635.2B	Protection of Classified National Security Council Information
75.	5635.3	Hand-Carrying Classified Matter on Air Carriers
76.	5635.4	Protection of Unclassified Controlled Nuclear Information
77.	5636.1	Prohibitions on Wiretapping and Eavesdropping
78.	5636.3A	Technical Surveillance Countermeasures Program
79.	5637.1	Classified Computer Security Program
80.	5650.3	Identification of Unclassified Controlled Nuclear Information
81.	5670.1	Management and Control of Foreign Intelligence
82.	5670.2	Security of Foreign Intelligence Information Facilities
83.	5700.6B	Quality Assurance
84.	5820.2A	Radioactive Waste Management
85.	6430.1A	General Design Criteria

3 QUALITY ASSURANCE

3.1 Background

In the past decade, Savannah River Site (SRS) processes to implement a quality assurance (QA) program achieved less than satisfactory results. In December 1981, the U.S. Department of Energy (DOE) provided a better focus for their QA requirements by invoking American Nuclear Standard Institute/American Society for Mechanical Engineers (ANSI/ASME) Standard NQA-1 as their desired standard (SR Order 5700.6A, Reference 1). However, the Savannah River Plant (SRP) QA Manual, DPW 83-111-3 (Reference 2), applicable to reactors, was not fully responsive to the DOE order and was never fully implemented. On April 1, 1989, Westinghouse Savannah River Company (WSRC) assumed the management of SRS and issued WSRC Quality Assurance Plan, WSRC-1-05, Revision 0 (Reference 3). The plan was approved by the Savannah River Operations Office on April 25, 1989.

WSRC created the Environment, Safety, Health and QA (ESH&QA) Division and assigned its Vice-President and General Manager with the overall QA responsibility. To ensure a unified approach to QA, the four existing QA manuals were replaced by a single manual, WSRC-1Q (Reference 4) for sitewide application. Both the plan and the manual reflect the requirements of the latest revision of DOE Order 5700.6B, "Quality Assurance" (Reference 5).

Total compliance with DOE Order 5700.6B (and WSRC-1Q) is planned to be achieved in two stages. In the Reactor Operations Management Plan (ROMP), WSRC-RP-89-368 (Reference 6), WSRC identified 41 QA Restart Work Items (RWIs) which, upon completion, will provide a significant upgrade of existing QA programs. The Reactor Safety Improvement Plan (RSIP) (Reference 7) addresses the longer term continuing effort towards total compliance by September 30, 1991.

3.2 Restart Criteria

The criteria for determining the adequacy of the SRS QA program for restart are described below.

- (1) Activities performed for the fulfillment of both Charlotte and Westinghouse Independent Safety Review (WISR) restart work items contained in the ROMP shall be accomplished in accordance with the requirements of SRP QA Manual, DPW 83-111-3, Revision 11, as specified in Volume 1, Section 2.4.4 of the ROMP.
- (2) All other restart activities shall be accomplished in accordance with the existing implementing procedures.
- (3) The 41 QA related RWIs identified in the ROMP shall be completed to provide a significant upgrade of existing QA programs and shall be in full compliance with NQA-1-1986.

- (4) The status of compliance with ANSI/ASME NQA-1-1986 shall be determined and a mutually agreed upon schedule of the remaining work to be done(subsequent to reactor restart) under the RSIP program for total compliance with DOE Order 5700.6B, shall be submitted to DOE.

3.3 Evaluation

Use of the new QA Manual, WSRC-1Q, is being progressively implemented within the Reactor Restart Division (RRD). However, it will not be fully implemented prior to restart of K-Reactor. This fact was recognized early on by WSRC, which put in place a specific restart QA program as described in the ROMP. The four restart criteria reflect the essence of this approach. Full implementation of QA Manual WSRC-1Q will be achieved under the RSIP program. The SRP QA Manual, DPW-83-111-3, will remain as the source document for verifications performed to support closure of the Charlotte and WISR restart work items (RWI).

Restart/RSIP Quality Project Teams (RQPT) have been formed and report to the Section Manager of Restart Quality Assurance within the Reactor Restart Division (RRD). The purpose of the RQPT is to develop the necessary procedures for the QA implementation throughout RRD, to assist line organizations in implementing the QA requirements, and to review and verify closure of restart work items.

Restart Criterion 1

The Reactor Restart Verification (RRV) surveillance program ensures that Charlotte and WISR RWIs are accomplished in accordance with the requirements of SRP QA Manual DPW 83-111-3. Procedure RQPT-I-004 (Reference 8) implements this program. The essential elements of the program are as follows:

- o verification plans based on SRP QA Manual DPW 83-111-3
- o performance of both interim and closure surveillances
- o identification and followup of deviations
- o verification of corrective action
- o close-out of surveillance reports

The techniques to be used for conducting surveillances are as follows:

- o reviewing, on a step-by-step basis, all applicable requirements with cognizant personnel
- o observing a sequence of events to determine if the activity is being performed in accordance with prescribed procedures (performance-based inspection)
- o review and evaluation of each characteristic or attribute to determine if the activity was or is being performed in accordance

with prescribed procedures or meets the criteria as stated in the deliverable scope

- o documenting evaluation of each characteristic/attribute listed in the verification plan, regardless of the result

Completed surveillance reports are transmitted to reactor restart files as part of the QA Closure Package for restart. As an added measure, the Safety Oversight Division (SOD) also reviews the closure process of Charlotte and WISR restart items on a sample basis.

SRP QA Manual DPW-83-111-3 is the principal source document for this program. However, the QA program upgrade provided by completion of the 41 ROMP QA restart work items (Criterion 3) is taken advantage of on a forward-fit basis (i.e., work already completed will not be backfitted to the revised QA program prior to restart).

The staff determined that the program in place at SRS is adequate to satisfy Restart Criterion 1.

Restart Criterion 2

Both the Charlotte Criteria and the WISR identify activities that have direct impact on safe operations of the reactor. All other restart activities are to be carried out in accordance with the existing implementing procedures. The thrust of this criterion is that, at the very least, activities must be carried out in accordance with written procedures applicable at that time. This is ensured by the surveillance program put in place by the Reactor Quality Verification Group (RQV).

Reactor Quality Standard Procedure (RQSP) 18-1 (Reference 9) defines this program. The program is similar to those established by the commercial nuclear power industry and is performance-based (i.e., it places emphasis on the observation and evaluation of activities-in-progress). The surveillance areas are as follows:

- o Reactor Operations
- o Reactor Engineering
- o Reactor Outage and Maintenance
- o Reactor Quality
- o Planning, Scheduling and Cost Control
- o Site Engineering and Services Quality
- o Savannah River Construction
- o Reactor Training and Procedures

The following factors are considered in determining surveillance frequency:

- o potential impact on nuclear safety and reliability
- o technical complexity of the activity
- o Frequency at which the activity occurs
- o past quality history and the activity
- o need to observe new programs with no past performance history

The RQV periodically reviews the Surveillance Report Log to assure that corrective action responses are received on time, and closes out surveillance reports after verification of corrective actions. A copy of the closed-out surveillance report is then transferred to Central Files.

The RQV Surveillance Program has been providing valuable support to the Reactor Restart Division over the past months by ensuring that safety significant problems are either prevented or detected early and resolved in a timely manner. Additionally, the Safety Oversight Division conducts day-to-day inspections and has so far looked into areas such as:

- o Nonconformance Reporting (NCR) Program
- o Training
- o Housekeeping
- o Maintenance
- o Reactor Quality Surveillance Program
- o Control of Special Process
- o Procurement
- o Control of Vendor Technical Manuals
- o Control of Measuring and Test Equipment
- o Control of Temporary Procedure Changes
- o Storage of Reactor System Spare Parts

The SOD tracks all findings and verifies corrective actions prior to closeout.

The staff determined that the program in place at SRS is adequate to satisfy Restart Criterion 2.

Restart Criterion 3

The 41 QA-related RWIs included in the ROMP were established to provide a significant upgrade of the existing level of QA program implementation. Procedure RQPT-I-003 (Reference 10) defines the closure process of these 41 RWIs. The RRV Surveillance Program is used to ensure that acceptance criteria have been satisfied and deliverables are complete.

The format of the closure package is as follows:

- o cover sheet defining the subject matter
- o Table of Contents
- o final approved documents

The following is a list of required documents to be included in the closure package:

- o cover letter
- o RRV Surveillance Report
- o deliverables
- o Resolution of Comments document, if any
- o supporting documentation

The RQPT Manager reviews and signs the cover letter to close action on closure packages. The Safety Oversight Division reviews all completed closure packages.

To ensure compliance with ANSI/ASME NQA-1-1986 (WSRC-1Q) the Restart QA Group has prepared a matrix linking each of the 41 RWIs to specific paragraphs of the standard. Each closure package, is reviewed against these specific requirements. The cover letter, which is signed by the RQPT Manager, contains a statement that the QA-program upgrade achieved is in compliance with ANSI/ASME NQA-1-1986 requirements.

The staff determined that the program in place at SRS is adequate, however, Restart Criterion 3 will remain an open item pending satisfactory closure of the 41 QA related RWI's.

Restart Criterion 4

The SRS Quality Assurance Status Report, ESH-QA-90081, dated September 1990 (Reference 11) provides current status of compliance with regard to WSRC-1Q Manual (based on ANSI/ASME NQA-1-1986). The extent of compliance is expressed in two categories: "Systems In-Place" and "Full Implementation." "Systems In-Place" is defined as when the administrative implementing procedures are in place and a schedule for implementing the procedures has been established. "Full Implementation" is accomplished when personnel are applying the implementing procedures on an operational level to all items/activities. The numbers indicating implementation levels are approximate and provide reasonable estimates. The current (as of December 1990) implementation levels of the various departments within RRD range from 55 percent to 80 percent for systems-in-place level and from 45 percent to 70 percent for full-implementation level. The status of individual departments are as follows:

o Reactor Operations

Fourteen sections of the WSRC-1Q Manual have requirements that are applicable to this organization. The current systems-in-place level for the organization is 60 percent, and the full-implementation level is 50 percent.

o Reactor Maintenance

Fifteen sections of the WSRC-1Q Manual have requirements that are applicable to this organization. The current systems-in-place level for the organization is 65 percent, and the full-implementation level is 45 percent.

o Reactor Training and Procedures

Seven sections of the WSRC-1Q Manual have requirements that are applicable to this organization. The current systems-in-place level for the organization is 65 percent, and the full-implementation level is 60 percent.

o Reactor QA and Assessments

Twenty sections of the WSRC-1Q Manual have requirements that are applicable to this organization. The current systems- in-place level for the organization is 60 percent, and the full-implementation level is 55 percent.

o Reactor Engineering

Sixteen sections of the WSRC-1Q Manual have requirements that are applicable to this organization. The current systems- in-place level for the organization is 80 percent, and the full-implementation level is 60 percent.

o Reactor Startup and Testing

Eleven sections of the WSRC-1Q Manual have requirements that are applicable to this organization. The current systems-in-place level for the organization is 85 percent, and the full-implementation level is 70 percent.

o Strategic Programs

Seven sections of the WSRC-1Q Manual have requirements that are applicable to this organization. The current systems- in-place level for the organization is 65 percent, and the full-implementation level is 60 percent.

o Planning, Scheduling and Cost Control

Seven sections of the WSRC-1Q Manual have requirements that are applicable to this organization. The current systems- in-place level for the organization is 55 percent, and the full-implementation level is 55 percent.

o Technical Programs

Eight sections of the WSRC-1Q Manual have requirements that are applicable to this organization. The current systems- in-place level for the organization is 60 percent, and the full-implementation level is 55 percent.

o Configuration Management

Seven sections of the WSRC-1Q Manual have requirements that are applicable to this organization. The current systems- in-place level for the organization is 90 percent, and the full-implementation level is 65 percent.

o Overall Levels

The overall systems-in-place level for RRD is 69 percent, and the full-implementation level is 58 percent.

RSIP schedules, both Level 1 for RRD's overall implementation and Level 2, for RRD's individual department's implementation, have been submitted to DOE via the "Compliance Schedule Approval Request - DOE/SR Order 5480.6" (Reference 12). These schedules reflect a significant effort on the part of RRD and WSRC to implement the QA program concurrent with the overall reactor restart program. The tasks completed through these schedules will build upon RRD's procedural base and result in revisions of existing administrative systems to reflect current organizational functions and responsibilities and to address all applicable requirements in the WSRC-1Q Manual.

The QA program elements that will require the most developmental effort are design control, records management, and software QA. The RSIP initiative began in July 1990 and has a completion date of September 30, 1991.

Closure of all ROMP RWIs ensures that adequate compensatory measures will be in place with regard to the 13 Designated Safety Systems listed in Attachment 1 of this SER section. The staff determined that the schedule submitted is acceptable and that Restart Criterion 4 has been satisfied.

3.4 Conclusion

The staff evaluated the implementation of the QA program within RRD and determined that it is being accomplished through three initiatives: (1) independent verification of Charlotte and WISR RWI closures, (2) completion and independent verification of 41 QA RWIs and (3) implementation of the WSRC-1Q Manual under the RSIP program. On the basis of its evaluation, the staff concludes that resolution of the open item identified above will satisfy the restart criteria for this issue.

3.5 References

1. WSRC, SR Order 5700.6A, "Quality Assurance," December 30, 1981.
2. WSRC, "SRP Quality Assurance Manual," DPW 83-111-3, Revision 11.
3. WSRC, "WSRC Quality Assurance Plan," WSRC-1-05, Revision 0.
4. WSRC, "WSRC Quality Assurance Manual," WSRC-1Q, Revision 0.
5. DOE Order 5700.6B, "Quality Assurance," September 23, 1986, including Change 1 dated March 28, 1990.
6. WSRC, "Reactor Operations Management Plan," WSRC-RP-89-368.
7. WSRC, "Reactor Safety Improvement Program Scopes and Schedules for Top 50 Ranked Improvements," WSRC-RP-89-384.

8. WSRC, "Reactor Restart Verification Surveillances," RQPT-I-004, Revision 0.
9. WSRC, "Surveillances," RQSP 18-1, Revision 0.
10. WSRC, "Control of RQPT Closure Packages," RQPT-I-003, Revision 0.
11. WSRC, "Quality Assurance Status Report," ESH-QA-90081, dated September 1990.
12. WSRC, "Compliance Schedule Approval Request - Draft DOE/SR Order 5480.6," ESH-QA-90055, June 19, 1990.

DESIGNATED SAFETY SYSTEMS

1. SCRAM (including safety rods)
2. Control Rod System
3. Supplementary Safety System
4. Central Control Room Habitability
5. Emergency Power
6. 125 V DC Power
7. Process Water Diesel Generators
8. Process Water System
9. Emergency Cooling System
10. Cooling Water System
11. Reactor Room Spray System
12. Discharge Assembly Cooling
13. Water Removal and Storage

4 RADIATION PROTECTION

4.1 Background

The purpose of this section is to review the radiation protection practices in use at the Savannah River Site (SRS) K-Reactor facility and to determine their acceptability when compared to DOE requirements and Westinghouse Savannah River Company (WSRC) corporate commitments.

It must be acknowledged that the broad changes underway in the site Health Protection program are independent of the restart effort. Several outside stimuli that predate the restart effort, such as issuance of DOE Order 5480.11, "Radiation Protection For Occupational Workers" (Reference 1), and the broad-scope assessment of the site Health Physics program performed by the Office of the Deputy Assistant Secretary for Safety, Health and Quality Assurance, DOE-HQ (Reference 2), have provided motivation for sweeping changes throughout the radiation protection program.

The goal of the programmatic changes that were already underway was to elevate the program to a higher standard of excellence. WSRC has developed a schedule (Reference 3) that will fully implement all elements of DOE Order 5480.11. While certain major capital projects will require several years to complete, the major portion of the plan is scheduled to be in place before the restart of K-reactor. Additionally, a program improvement plan (Reference 4) has been developed to set the course for future program enhancements.

4.2 Restart Criteria

The criteria for determining the adequacy of SRS radiation protection practices for restart are described below.

(1) Compliance with DOE Orders

DOE Orders are issued to provide reasonable assurance that operation of the facility in the manner indicated will not endanger the environment, the health and safety of the public, or the health and safety of DOE or contractor employees. WSRC shall fully comply with all applicable DOE Orders relating to Radiation Protection/Health Physics unless alternative means have been demonstrated and have been approved by DOE that provide an equivalent means of achieving the required margins of safety. DOE orders which are applicable to the Radiation Protection Program are listed below.

- o DOE Order 5480.11, "Radiation Protection for Occupational Workers."
- o DOE Order 5480.4, "Environmental Protection, Safety and Health Protection Standards" (Reference 5), which prescribes the application of certain radiation protection standards at DOE facilities.

- o DOE Order 5480.6, "Safety of DOE Owned Nuclear Reactors," Section 8.e, "Reactor Personnel Training and Qualification Program" (Reference 6), which prescribes the standards for training and qualifications of Health Protection Technicians at DOE reactor facilities.
 - o DOE Order 5820.2A, "Radioactive Waste Management" (Reference 7); Chapter III, "Management of Low-Level Waste"; Section 3.c, "Requirements for Waste Generation," which requires technical and administrative controls to be directed to reducing the gross volume of waste generated and/or the amount of radioactivity requiring disposal.
- (2) WSRC shall develop and implement a plan (Reference 8) to resolve the elements of the 1988 SRP K-Reactor Restart Strategy (Reference 9). Specifically that plan shall address the following elements:
- (a) Strict compliance with existing requirements shall be enforced.
 - (b) A policy shall be developed and implemented which establishes that compliance with radiation protection requirements is the ultimate responsibility of the Reactor Program Plant Manager.
 - (c) A management audit and inspection program is established which ensures compliance with radiation protection procedures and requirements and assesses overall plant performance.
- (3) WSRC shall complete the action plan (Reference 10) developed to correct the root causes identified in the January 23, 1989, DOE review of SRS Health Physics.

4.3 Evaluation

Compliance with DOE Orders

Compliance with DOE Order 5480.11, "Radiation Protection for Occupational Workers"

In June of 1989, the WSRC Environmental & Health Protection Department (EHP), now known as Health Protection Department (HPD), presented DOE with a draft document that provided an internal assessment of WSRC compliance with DOE Order 5480.11. Areas of noncompliance were accompanied by a list of actions and a schedule to achieve full compliance. Compliance with the Order has generally progressed according to WSRC's implementation plan, with some noteworthy delays and postponements. The following evaluation corresponds to individual elements of the Order.

- o Maintaining Radiation Exposures As Low As Reasonably Achievable (ALARA)

WSRC's implementation plan indicated that the ALARA program for SRS complies with the Order, which requires that plans and programs used to ensure that occupational radiation exposures are maintained ALARA shall be documented. The SRS ALARA policy is documented in Procedure Manual WSRC-1-01, "Management Policy 4.4, Radiation Protection" (Reference 11). Documentation of the site's ALARA program is provided in Special Hazards Bulletin 9 (SHB-9), "Reducing Radiation Exposure to ALARA at the Savannah River Plant" (Reference 12). In addition, the SRS ALARA Guide (Reference 13), which was issued on December 14, 1990, describes key elements of the SRS ALARA program. These documents reflect current practices presented in DOE publication PNL-6577, "Health Physics Manual of Good Practices for Reducing Radiation Exposure to Levels that are ALARA" (Reference 14). This publication is recommended by the Order as a source of useful practices for achieving the objective of the ALARA process.

A January 23, 1989, appraisal of the ALARA program pointed out that, although the average employee occupation radiation exposure for SRS workers was relatively low compared to the commercial nuclear industry, there were certain weaknesses in the program. It was noted that poor housekeeping practices and certain physical and administrative controls for maintaining exposures ALARA did not meet current standards. Subsequent surveillances by DOE staff and management have indicated a long-term positive trend in the housekeeping practices. Other improvements such as the policies described in the paragraph above have addressed a number of the weaknesses in the administrative controls.

The Radiation Work Permit (RWP) Program (Reference 15) is continuing to be an integral portion of the ALARA program by improving job planning and ALARA reviews performed prior to work in radiation areas.

In summary, the DOE review of all programs in place indicated that the contractor was in compliance with the requirements of this section of the Order.

o Radiation Protection Standards for Internal and External Exposure for Occupational Workers

WSRC's assessment indicated that it was in compliance with this element except in the area of nonstochastic effects of radiation exposure. WSRC reported in May 1990 (Reference 16) that subsequent changes to procedures and employee training programs eliminated all areas of noncompliance. A DOE review of programs and procedures such as Special Hazards Bulletin 7 (SHB-7), "Radiation Exposure Control" (Reference 17), indicated that WSRC was in compliance with the requirements of this section of the Order.

o Planned Special Exposure

The contractor reported that it was in compliance with this element after changes were made to SHB-7. A review of SHB-7 indicated that WSRC is in compliance with the requirements of this section of the Order. As an enhancement, WSRC will revise SHB-7 to clarify the level of authorization required to approve a second planned exposure to the same individual.

o Radiation Protection Standards for Internal and External Exposure to Minors and Students

The contractor reported that it was in compliance with this element after changes were made to Section I,C,1.1 of DPSOP-40, "SRS Radiation and Contamination Control" (Reference 18); DPSOL-193-220, "Health Protection Requirements for Visitors" (Reference 19); and Section 104.02 of SHB-7. One of the changes was to add a mechanism to identify minors. A review by DOE of the listed procedures did not locate the mechanism. This will remain an open item.

o Radiation Protection Standards for Public Entering a Controlled Area

The contractor reported that changes to SHB-7 as well as employee training programs, eliminated all areas of noncompliance. A review by DOE of Section 104.02 of SHB-7 indicated that WSRC is in compliance with the requirements of this section of the Order. However, there appears to be weaknesses in this program involving positive control of access to RCAs. (Refer to comments under "Entry Control Program," for more discussion.) This will remain an open item until the discrepancy is resolved.

o Procedural Requirements

The contractor reported that it was in compliance with procedural requirements after four new procedures were issued and changes were made to "WSRC Management Requirements and Procedures" (Sections 4.07 - 4.10) (Reference 20); DPSOP-40, "SRS Radiation and Contamination Controls"; DPSOP 193-203, "Handling Contaminated Cases" (Reference 21). The newly issued procedures are:

DPSOL 45-379, "Evaluation of Non-Uniform Radiation Exposure To Skin" (Reference 22),

DPSOL 47-1-501, "Bioassay Follow-Up and Intake Assessment" (Reference 23),

DPSOL 47-1-502, "Calculation of Internal Dose" (Reference 24)

DPSOL 47-1-503, "Internal Dosimetry Dose Assessment Records and Reports" (Reference 25)

The DOE review of the programs in place indicated that the contractor is in compliance with the requirements of this section of the Order.

o Monitoring

The Order requires monitoring of both employees and their workplace to demonstrate compliance with radiation protection standards and to identify potential sources of exposure. The contractor's internal assessment indicated only partial compliance in each of four categories. An assessment of actions taken to-date to achieve compliance and the remaining areas of noncompliance follows.

External Radiation -- Procedures were to be revised to establish the new threshold dose values for which personnel dosimeters would be required. Changes were made to Special Hazards Bulletin 7 (SHB-7), "Radiation Exposure Control," which satisfied the intent of this requirement. In order to ensure that radiation workers subject to annual external radiation exposures greater than 100 millirem are monitored for such exposure, WSRC chose to require all workers who enter the reactor's protected area fence to wear thermoluminescence dosimeters (TLD) badges. The General Employee Training (GET) and Radiation Worker Training (RWT) programs instruct employees of this requirement and entrances to the protected areas are posted as such. However, there is still concern over the lack of a plan to ensure worker compliance. (See "Entry Control Program," below for more discussion.)

There also appears to be a weakness in the process used to follow through on lost or unreturned TLD badges. Presently, procedures require an investigation into such events only after 6 months has elapsed.

An assessment of the personnel dosimetry program by a joint team from the DOE-Savannah River Special Projects Office (SRSPPO) and DOE-Savannah River Operations (SR) Office is scheduled during January 1991. This will remain an open item until the results of the assessment are known and any discrepancies are resolved.

Internal Radiation -- The contractor's internal assessment revealed the new criteria for an internal dosimetry program were not stated in plant documentation. Additionally, a method or procedure to identify workers who meet the criteria did not exist, and a complete evaluation of all contributions to internal dose was required.

The contractor has reported that actions required to achieve full compliance in the reactor facilities have been completed including modification of procedures and computer programs for dose calculations. Additionally, WSRC has issued a Technical Basis Manual (Reference 26) intended to provide a technical and philosophical discussion of the radiobioassay and dose assessment aspects of the internal dosimetry program.

The contractor has reported that the internal dose evaluation program for K-Reactor incorporates extensive bioassay sampling for tritium and state-of-the-art whole body counting for fission products with Fast Scan devices. The contractor has stated that it is confident that both the tritium bioassay program and the Fast Scan whole body analysis meet the requirements of the Order.

An assessment of the personnel dosimetry program by a joint team from the DOE-SRSP0 and DOE-SR Operations Office is scheduled during January 1991. This will remain an open item until the results of the assessment are known and any discrepancies are resolved.

Workplace (Air Monitoring) -- The contractor reports compliance with requirements for air monitoring in the reactor areas. Programs have been underway to identify and quantify any radiological hazard due to airborne particulate matter. To date, no measurable hazard has been identified. Programs have also been underway to quantify the radiological hazard associated with airborne tritium.

The review of the results of this program has been completed by DOE. While the hazard due to airborne radioactive particulate matter was found to be seemingly insignificant, industry standards (Reference 27) and this section of the Order specifically require a timely evaluation of any possible hazard, regardless of any "baseline" study. Such baseline studies have averaged airborne concentrations over 7-day periods, and do not have the capability of evaluating transient or periodic events. Additionally, monitoring equipment presently in use does not have the required sensitivity to provide live-time monitoring of airborne tritium concentrations at the levels required by this Order (i.e., 10 percent of the DAC limit). More sensitive equipment is being evaluated, but the results are not yet available. This will remain an open item until programs, equipment, and techniques can be improved to meet the requirements of this section.

Workplace (Radiation Monitoring) -- The contractor has reported that it is in compliance with the requirements for radiation monitoring in the reactor areas. Contrary to this, DOE reviews (Reference 28) of the quality assurance program for measuring and test equipment used for radiation monitoring indicate significant weaknesses in the calibration program for radiation monitoring

instruments. Additionally, concerns have been raised (References 29 and 30) by the Environmental, Safety and Health Site Representative Office of DOE about the adequacy of the WSRC's documentation of a technical basis for radioactive material and contamination monitoring and control measures. Other concerns involving inadequacies in the radiation monitoring program relative to certain industry standards are expressed under the section on compliance with DOE Order 5480.4.

This concern has not yet been adequately addressed by the contractor. This will remain an open item until all program discrepancies are resolved.

Contamination Control and Monitoring -- Procedures were revised to incorporate new contamination limits. Standard methods and units for recording contamination levels were incorporated into revised procedures. Upgrades in personnel monitoring equipment have been completed within the reactor facility, including installation of portal monitors at the facility perimeter. In spite of the significant improvements mentioned, major program deficiencies appear to remain. Specifically, items such as the concern raised by the EH-Site Rep and the apparent absence in the reactor facilities of routine tritium contamination surveys must be resolved. This will remain an open item until all program discrepancies are resolved.

o Methods of Estimating Dose Equivalent

Dose calculation methods have been formally documented in the Technical Basis Manual issued on December 20, 1990. However, an assessment of the personnel dosimetry program, including internal dose assessment methods, will not be completed by DOE until the end of January 1991. This will remain an open item until the results of the assessment are known and any discrepancies are resolved.

o Releases of Material and Equipment from Radiological Areas

The contractor's implementation plan indicated that existing procedures did not adequately detail the new requirements for releasing material to controlled or uncontrolled areas. The areas of noncompliance were to be corrected with procedural changes (Reference 31). While these changes were reviewed by DOE and found to be generally acceptable in prescribing adequate surveys on items to be released to controlled or uncontrolled areas, certain additional programmatic weaknesses remain. Specifically, the following weaknesses were noted:

- an inadequate quality assurance program for measuring and test equipment used for radiation monitoring

- the apparent absence of tritium contamination surveys in the reactor facilities
- poor documentation of a technical basis for radioactive material and contamination monitoring and control measures

This will remain an open item until all program discrepancies are resolved.

o Design and Control

The contractor's implementation plan indicated that SRS complies with the Order, which states that radiation exposure rates in controlled workplace areas should be reduced to ALARA levels by facility design and control. The Order states that the primary means are to be through physical controls with reliance on administrative and procedural means to be considered only as supplemental measures. After a review of DPSOP-40, a draft of the new Radiological Controls Manual - 5Q, Section RC-1001, "Design of New Nuclear Facilities" (Reference 32), and several memoranda (References 33-36) between and within WSRC's Health Protection Department and Engineering and Projects Division, DOE has concluded that the programs in place indicate that the contractor is in compliance with the requirements of this section of the Order.

o Posting and Labeling

The contractor reported that it was in full compliance with posting and labeling requirements in K-Reactor area in December of 1989 (Reference 37) after all radiological areas were reposted and procedures such as DPSOL 193-118, "Radiological Posting" (Reference 38), were revised to reflect the changes. Personnel having access to K-Reactor area were trained in the new posting requirements. Numerous DOE surveillances of K-Reactor confirmed that the reposting effort has been completed and that this requirement has been satisfied.

o Entry Control Program

The Order requires a measured approach to the control of personnel entry into radiological areas. Control may be as simple as a buffer zone between clean areas and radiological areas or as strict as locked or guarded entrances to very high radiation areas. The contractor's implementation plan called for a review of each entry location, upgraded postings and barricades and, if necessary, physical controls to limit access to high radiation areas. The contractor reported that it was in compliance with the requirements for an entry control program (Reference 39) after methods for controlling access to "High Radiation Areas" and locked "Very High Radiation Areas" were included in DPSOL-193-118. However, review by DOE of programs to identify visitors and to

limit their access to Radiologically Controlled Areas (RCAs), indicated that those programs did not appear strong enough to meet the requirements of the Order.

Access to RCAs in the reactor areas is controlled at several levels. Initially, access is controlled by passage through a security gatehouse that the contractor has chosen to use as the point past which personnel are to be equipped with a TLD for monitoring the annual effective dose equivalent from external irradiation. It is assumed, at this point, that possession of a TLD indicates authorization, for radiological control purposes, to enter the controlled area. This approach has a weakness. Wackenhut Services Incorporated (WSI), the contractor that provides security services, has not complied with a request from WSRC to assist them in providing screening for TLDs at the gatehouse. Subsequent RCA access controls include barriers and postings at entrances to each RCA. However, visitors who may have passed unimpeded through the security point would now be free to access a number of RCAs whose entrances are not controlled by Health Protection Inspectors. This will remain an open item until a demonstration of positive access control has been provided.

o Records

The contractor's implementation plan indicated that SRS complies with the Order for generation and maintenance of records for the following programs: ALARA, Monitoring and Area Control, and Monitoring Methods. Partial compliance was indicated in the training records program and the individual occupational dose records program involving internal dose. The contractor reported that internal dose data for reactor workers were available in existing records. It was also reported that training records existed in accordance with departmental training policies; however, they were not centrally located to permit ready retrieval. Completion of a commitment by the contractor to establish a program for the standardized documentation and retention of training records has been accomplished.

o Reports to Employees

The contractor's implementation plan indicated that SRS partially complies with the Order for this element. The contractor reported that the areas of noncompliance were corrected by changes to DPSOP-40; however, the ability to provide required exposure history reports to terminating employees within the 90-days still has not been achieved. This will remain an open item until the ability to produce such reports within the required timeframe is achieved.

o Radiation Safety Training

Radiation Safety Training - All Employees -- The contractor reported that the program did not ensure that orientation (or retraining) was provided to all employees who could enter a controlled area. Subsequent reports indicate that a revision of employee orientation materials has been completed. Training programs have recently been implemented to provide training to all new employees within 30 days of arrival on site. Additional programs are also in place to provide retraining to all incumbent employees who will have unescorted access to the production reactor facilities.

The DOE review of all programs in place indicated the contractor is in compliance with the requirements of this section of the Order.

Radiation Worker Training -- The contractor has reported that the program did not ensure that all radiation workers received adequate training or testing prior to or concurrent with assignment to radiation work. Subsequently, revisions to radiation worker training materials have been completed, and implementation of the new training/testing program is underway. Additional programs have also been established to provide retraining to all incumbent employees who will have unescorted access to Radiological Control Areas in the production reactor facilities. The DOE review of the programs in place indicated that the contractor is in compliance with the requirements of this section of the Order.

Health Protection (HP) Inspector Training -- The contractor has reported that the program did not include all topics required by the Order. Additionally, retraining every 2 years was not required. The Inspector retraining program was implemented in October of 1989. Revision of HP Inspector training materials was completed, and implementation of the new training program began. The DOE review of all programs in place indicated that the contractor is in compliance with the requirements of this section of the Order.

o Guidance and Requirements for Emergency Exposure During Rescue and Recovery Activities

The contractor reported that it was in compliance with this element after changes were made to Special Hazards Bulletin 7 (SHB-7), "Radiation Exposure Control," and procedure EPIP 6Q7-201, "Emergency Exposure Guidelines" (Reference 40), was issued. However, changes to training programs for emergency personnel have not been reviewed for adequacy. This will remain an open item until such programs and procedures can be reviewed by DOE.

o Nuclear Accident Dosimetry

The contractor's implementation plan indicated that SRS complies with the Order for nuclear accident dosimetry in the following portions of the program:

- All basic elements except for a formal procedure for analysis of Phosphorus-32 activity in hair. To resolve this, an approved procedure, "Procedure for Analysis of Phosphorus-32 Activation in Hair," (Reference 41) was added to DPSOL 45-1, "Evaluation of Personnel Exposures in the Event of a Nuclear Incident" (Reference 42).

A review by DOE of the program now in place indicated the contractor is in compliance with the requirements of this section of the Order.

- All elements of the Fixed Nuclear Accident Dosimeter program except for the following:
 - (a) Fixed units that are to be capable of determining neutron dose in rad (gray) with an accuracy of ± 25 percent. Numerous tests of the Criticality Neutron Dosimeter (CND) at the HPRR facility in the past have shown the badge provides sufficient accuracy for neutron dose determination. However, such tests were conducted with the CND mounted on a phantom (i.e., to collect on-body measurements). Insufficient tests have been performed to determine the accuracy of the units when not mounted on a phantom (i.e., to collect fixed unit measurements). Completion of this requirement is dependent on the availability of the Health Physics Research Reactor (HPRR) at Oak Ridge. No other facilities suitable for performing the required tests are known to the SRS staff. This will remain an open item until either the tests can be completed in a prompt manner or a suitable alternative is approved by DOE.
 - (b) Dose range capability of the gamma components of the units. Following the initial review, development of methodology and procedures for use of TLD readers to evaluate the gamma components of the dosimeters was completed and approved. Procedure 5Q2.1-378, "Reading High Range TLD 700 Chips" (Reference 43), was issued on December 21, 1989, to address this topic. A review by DOE of the program now in place indicated that the contractor is in compliance with the requirements of this section of the Order.

(c) Number and placement of dosimeters which are to be periodically re-evaluated. Additionally, an analysis that demonstrates that the dosimeters and their placement will satisfy the performance criteria is required to be documented. The development of a procedure for periodic re-evaluation of fixed Criticality Neutron Dosimeter (CND) unit placement was completed (DPSOP 133-215A). However, a complete technical assessment of the facility to ensure that fixed CNDs are in all required locations has not been completed. The contractor believes a full assessment of the scope indicated by a literal interpretation of the requirement may not be available before December 31, 1995. A less technically complex assessment can be completed by June 30, 1991. This will remain an open item until the assessment can be completed or a suitable alternative can be presented to and be approved by DOE.

- All elements of the Personnel Nuclear Accident Dosimeter program except procedures in place at the present time do not require personnel who occasionally enter areas requiring an installed criticality alarm system to wear a CND. The contractor states that compliance may require the purchase of approximately 10,000 additional dosimeters and the procurement of personnel and equipment to service them. Full compliance under this plan may require up to 3 years. The contractor believes several options exist that may comply with the requirements of the Order, including the request for an exemption. At this time, WSRC management has not decided on which option to pursue, but expects to do so within several weeks. This will remain an open item until the path forward can be determined and the program reviewed by DOE.

o Contractor Internal Audits

The contractor reported that it was in compliance with this element after changes were made to DPSOL-292-404, "Health Protection Department Internal Audits and Appraisals" (Reference 44). A review by DOE of all programs in place indicated that the contractor was in compliance with the requirements of this section of the Order.

Compliance with DOE Order 5480.4, "Environmental Protection, Safety and Health (ES&H) Protection Standards"

The Order contains three lists of standards. Attachment 1 addresses those standards that are mandatory as a result of non-DOE Federal or state ES&H statutes and/or implementing requirements. Under the topic of Radiation Protection, no standards are listed. Under the topic of Radioactive Material Packaging and Transportation Safety, two standards (References 45 and 46) from

the Code of Federal Regulations (CFRs) are listed. Attachment 2 addresses those standards that are mandatory as a matter of DOE policy. Under the topics of Radiation Protection, Industrial Hygiene, and Radioactive Material Packaging and Transportation Safety, eight standards (Reference 29 and References 47 through 53) from the American National Standards Institute (ANSI) and the International Atomic Energy Association (IAEA) are listed that may be applicable to the SRS production reactors. Attachment 3 addresses those standards and guidelines that are not mandatory but are provided as references on good practices and general informational purposes. Numerous codes, standards, and guides are listed which may be applicable.

The actual level of contractor compliance with the applicable requirements of the Order is unknown at this time. The WSRC ESH & QA - Related Order Compliance Plan (Reference 54) is incomplete; however, a detailed compliance assessment is underway. Based on disclosures from the contractor and reviews by DOE, it is known that full compliance is lacking with six of the ANSI standards listed in Attachment 2 of the Order. Specifically:

- ANSI N42.18-1980, "Specification and Performance of Onsite Instrumentation for Continuously Monitoring Radioactivity in Effluents"
- ANSI N319-1976, "Personnel Neutron Dosimeters (Neutron Energies Less Than 20 Mev)"
- ANSI N320-1979, "Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation"
- ANSI N322-1975, "Inspection and Test Specifications for Direct and Indirect Reading Quartz Fiber Pocket Dosimeters"
- ANSI N323-1983, "Radiation Protection Instrumentation Test and Calibration"
- ANSI-Z88.2-1980, "Practices for Respiratory Protection"

Compliance with DOE Order 5480.4 will remain an open item pending receipt of a WSRC report of full compliance and a review by DOE for adequacy or an adequate resolution between the contractor and DOE.

Compliance with DOE Order 5480.6, "Safety of DOE Owned Nuclear Reactors," Section 8.e, "Reactor Personnel Training and Qualification Program"

The upgraded Inspector retraining program was implemented in October of 1989. Revision of HP Inspector training materials was completed, and implementation of the new training program also began. While the HP Inspector training program is currently being upgraded to achieve DOE accreditation, it currently is in compliance with the requirements of the Order. Specific noncompliances that existed with respect to experience levels and qualifications of the Health Protection Inspectors has been resolved by implementing a qualification program. In addition to requiring specific minimum periods of time at each training grade, each inspector candidate must complete a qualification

check-off sheet documenting which job tasks the candidate has successfully demonstrated the ability to perform. DOE has concluded that with the programs in place indicate that the contractor is in compliance with the requirements of the Order.

Compliance with DOE Order 5820.2A, "Radioactive Waste Management"

Areas of concern involve programs to ensure minimization and segregation of radioactive wastes. While priorities involving the management of radioactive waste are established separately and not directly linked to the restart of K-Reactor, it should be realized that an effective ALARA program depends, in part, on an effective waste minimization and segregation program. To be effective, such a program must be in place prior to the operation of any facility that generates radioactive waste.

A WSRC self-assessment of compliance with this Order has not yet been made available for review. However, surveillances at the K-Reactor facility by DOE and interviews with representatives of the contractor have provided insight into the status of any such programs.

An effective waste minimization program should have at least three basic elements: (1) screening of materials brought into the RCAs in order to reduce the volume of material that may become contaminated, (2) survey of materials removed from RCAs to segregate the non-contaminated waste from the volume of contaminated waste, and (3) a method of reducing the volume of waste destined for disposal. A volume reduction program has been implemented; however, weaknesses in the material survey and control program must be resolved before it will be effective. Additionally, programs to restrict the introduction of potential waste generating material into RCAs has not yet received adequate management attention to be truly effective. This will remain an open item until such programs can be improved to the point deemed acceptable by DOE.

4.3.2 Completion of the Action Plan to Resolve Elements of the K-Reactor Restart Strategy

Enforce Strict Compliance with Existing Requirements (Issue MS 11.1)

The action plan to resolve this issue consisted of four deliverable items:

Upgrade Equipment -- The scope involved the procurement and installation of new whole body friskers and portal monitors. Installation of the whole body friskers and portal monitors has been completed. It should be noted; however, that questions discussed in Section 4.3.1 concerning the adequacy of the contractor's documentation of a technical basis for the use of this equipment must still be addressed. This will remain an open item.

Increase Staff -- The scope called for an increase in the minimum HP staffing in four categories. The Inspector staffing would be increased to four per shift on evening and midnight shifts during reactor outages and two per shift during reactor operation. Additional HP Inspectors would be provided on the day shift crew, as needed, to cover periods of

high work activity in radiologically controlled areas. At least one HP supervisor would be placed on each shift. At least one technical person with prior radiation protection experience would be added to the reactor HP staff. The staff determined that the reactor HP staff was increased as required.

Revise HP Administrative Controls -- The scope involved establishing transition points (TPs) at major entrances to posted radiological areas. The TPs would be manned by an HP Inspector who would be responsible for controlling access to the area and radiological activities within the area. The TP program was replaced in May of 1990 with the Control Point Program, which has proven more efficient and productive. For the purpose of complying with this commitment, this change to the Control Point Program is acceptable.

Surveillance Reports -- Reactor Quality department would perform surveillances of compliance to ensure that deliverables identified under this criteria are capable of being implemented and that personnel are complying. Deviations identified during the surveillances would be resolved and findings closed prior to restart. A schedule for performance of the surveillances was published, and completion of the surveillances was reported. On the basis of a review of the surveillances and their associated responses, the DOE staff has determined that the program was functioning adequately and indicated the contractor has met the requirements of this commitment.

A Policy Shall be Developed and Implemented Which Establishes that Compliance with Radiation Protection Requirements is the Ultimate Responsibility of the Reactor Program Plant Manager (Issue MS 11.2)

The action plan to resolve this issue consisted of one deliverable item: establish a policy that compliance with radiation protection requirements is the responsibility of the RPMT line management (the immediate responsibility of the K-Reactor Plant Manager). The approach (Reference 55) was to revise the reporting lines of HP Inspectors (see Section 4.3.3). Reference was made to close this item in conjunction with MS-12.1, "Management Structure - Accountability." This calls for establishing an effective management structure by clearly defining line management accountability.

Establish an Audit Function to Sample Performance (Issue MS 11.3)

The action plan to resolve this issue consisted of one deliverable item: continue MS-11.1 scope until performance dictates a 50 percent reduction of frequency of deviations from first week of implementation (based on weekly average). One inspector assigned to each shift will audit monitoring equipment and its proper usage at TPs not normally manned by an HP Inspector. An HP technical person will audit compliance of all radiation workers, provide training for HP and other personnel, and provide assistance in planning of work in the radiologically controlled areas.

The contractor reported in their Issue Closure Package (Reference 56) for topic MS-11, "Health Protection" (dated May 3, 1990), that the requirements of MS-11 had been met. In addition, the contractor provided surveillance reports to substantiate the claim. Further review by DOE indicated the program was functioning adequately and indicated the contractor had met the requirements of this commitment.

Completion of the Action Plan to Resolve the Following Concerns of the DOE/HQ Health Physics Review of the SRS (January 23, 1989)

In response to the review, the Manager of the Health Protection Department (HPD, formerly known as the Environmental and Health Protection Department [EHP]) stated that HPD would correct the root causes identified in the review to strengthen their program. The first four root causes were then addressed and a response provided. Listed below is a brief description of the contractor's response and a summation of any activities performed to meet the commitment.

o Lack of Effective Management Controls

The response to this element included the intent to establish a multilevel oversight program to monitor and ensure procedural compliance. Listed below are the stated goals of that program and an assessment of the status of each goal as of January 1991.

Internal Appraisals -- Appraisals of the SRS HP program would be performed by the Westinghouse Government Operations Nuclear Safety and Environmental Oversight Committee. Appraisals by this committee were to have begun during October 1989 and be repeated at least annually. Since this commitment was made, the committee has made two visits to SRS. A study of the reports from these visits failed to reveal any review by the committee of radiation protection activities at SRS. There was a brief acknowledgement in the first report of reduction of tritium releases and personnel exposure at the site Tritium Facility; however, it would not qualify as a review of the program. It would appear, based on a review of the reports, that the objective of the committee is to provide an objective review of much broader areas than singular programs, such as radiation protection, and DOE does not expect that the committee's evaluation would provide the required depth. On the basis of its review, the DOE does not expect that this commitment will be an effective tool for assessing the program. This will remain an open item pending resolution of this concern.

Audits by GOCO Health Physicists -- Audits performed by a committee comprised of health physicists from various other Westinghouse-operated Government Owned/Contractor-Operated (W-GOCO) sites were to begin in 1989. By the end of 1990, no formal audits had taken place. A schedule for the W-GOCO HP audit in 1991 has been established for the first quarter of the year. This will remain an open item pending completion of the scheduled audit.

Reviews of Radiation Protection Activities by INPO -- INPO (Institute of Nuclear Power Operations) was to be requested to schedule an expanded review of radiation protection activities in the fourth quarter of 1989. Due to major programmatic changes underway during the fourth quarter of 1989, the request for an assistance visit was delayed until after the changes could be completed. The INPO special assistance visit was completed during the fourth quarter of 1990, and the results forwarded to both WSRC and DOE. A review by DOE indicates that the contractor has met the requirements of this commitment.

Internal Audits --HPD would increase the number of health physicists assigned solely to internal audits. Although DOE Order 5480.11 requires internal audits of all the functional elements of the radiation protection program at least every 3 years, WSRC committed in a memorandum to DOE that all elements would be audited annually starting in 1990. Three additional staff positions were requested; however, after filling one position, budget constraints eliminated the two additional positions. In addition, expanded staff responsibilities necessitated reassigning one of the staff members to non-audit functions. Contrary to the commitment to increase the audit staff, as of January 1991, the staff size remains constant. Due to the lack of progress of assigning adequate resources to this program, this will remain an open item until changes are made which support the commitment.

Experienced Quality Assurance (QA) personnel would be added to the HPD staff to audit QA compliance of Health Protection. In addition, QA performance would be audited by the Environment, Safety, Health and Quality Assurance (ESH&QA) Division's QA Section. During 1989, the contractor was successful in adding one employee with QA experience to the staff. The QA Section of ESH&QA has continued to perform audits of HP practices during 1990. The review by DOE indicated that the contractor had met the requirements of this commitment.

Experienced health physicists, including some with commercial nuclear experience, would be added to the Health Protection groups in the operating areas. The total number of degreed health physicists assigned to the three reactor areas has increased from one in 1988 to four by the end of January 1991. Three of the four also have commercial nuclear power experience. Additionally, the position of health protection manager for K- and L-Reactors were filled with a degreed health physicist with commercial nuclear experience. The review by DOE indicated the contractor had met the requirements of this commitment.

The Health Protection work force would be substantially increased to allow stationing additional inspectors at major entrances to radiation areas to audit conformance with radiation protection requirements. To meet this commitment, the Health Protection Inspector staff at each reactor was tripled from approximately 10 to 30 Inspectors. The review by DOE indicated the contractor had met the requirements of this commitment.

Lack of Line Management Acceptance of Responsibility

The contractor issued a management policy assigning the responsibility to line managers for ensuring that their operations are conducted safely and in accordance with applicable radiological control procedures and ALARA principles. Additionally, organizational changes are being made in the reporting path of the facility HP staff that will result in the plant manager of K-Reactor being directly responsible for all radiological protection activities at that facility (Reference 55). HPD, which will retain technical direction of the facility HP staff, also intends to audit the acceptance of this responsibility by line management by intensely monitoring certain key indicators. Presently, the changes in the reporting path of the facility HP staff have been completed, but DOE is still assessing their acceptability.

Nonconformance of SRS Health Physics Requirements

HPD formed a team of senior health physicists with power reactor, INPO, and SRS experience to identify and initiate near-term upgrades in radiation protection practices and procedures. Their efforts resulted in the issue of the "WSRC Radiation Protection Improvement Program," as presented August 28, 1989, by J. S. Moore, President, WSRC. Implementation of the improvement plan is progressing adequately and will benefit the total radiation protection program. The improvement plan is intended to be a boundless program that will continually seek out areas for improvement; therefore, completion of the existing plan is not considered a prerequisite for reactor restart.

Insufficient Attention to Human Performance

SRS addressed human performance through the implementation of expanded training and requalification programs. The Inspector Training Program would be improved to meet the requirements of DOE Order 5480.11 and other regulations. Inspector retraining would be implemented for the first time at SRS. The Health Protection Training Group would be expanded to support the increased training demands and would be staffed, in part, with employees who have commercial nuclear experience. Considerable enhancements in training programs for radiation workers was also scheduled.

The Training Group was increased in size. Additionally, the position of Group Leader was elevated to that of Section Manager and filled by an individual who possessed previous experience as an HP training manager with a commercial nuclear utility. Inspector retraining was implemented on a rotating basis to ensure retraining of all HP Inspectors within 2 years. Improved lesson plans for radiation worker training was prepared. A major weakness still exists in that no dedicated facility exists for radiation worker or Health Protection Department training; however, the program is functioning in agreement with the commitment. The review by DOE indicated the contractor had met the requirements of this commitment.

4.4 Conclusion

As a result of this evaluation, it is apparent that a number of major improvements have been made in the Radiation Protection program. Not only are major program areas being upgraded to the more stringent requirements of the DOE Order 5480.11, but significant progress continues to be made toward achieving a program based on a standard of excellence. While progress is being made, the Radiation Protection program does not yet meet the restart criteria for restart of the K-Reactor facility. The Radiation Protection program will remain an open issue until all areas of noncompliance with the listed restart criteria are closed based on either verified compliance or an adequate resolution between the contractor and DOE. The items closed by resolution between WSRC and DOE, which involve compliance with DOE orders, will be done so in accordance with DOE Memorandum, "Procedure for Requesting U.S. Department of Energy (DOE), Savannah River Operations Office (SR) Directive Compliance Schedule Approvals (CSAs) and Exemption Approvals" (January 18, 1990) (Reference 57).

4.5 References

1. DOE Order 5480.11, "Radiation Protection For Occupational Workers" (December 21, 1988 and Change 1, July 20, 1989).
2. Kenneth P. Ferlic, et al., DAS/HSQA DOE-HQ, "Savannah River Plant Health Physics Review," January 23, 1989.
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15. WSRC, DPSOL-292-429, "Radiation Work Permit," September 8, 1989.
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32. Radiological Controls Manual - 5Q, Section RC-1001, "Design of New Nuclear Facilities," (Draft).
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5 DESIGN CONTROL AND VERIFICATION

5.1 Safety Analysis Report

5.1.1 Background

The original (1972) Safety Analysis Report (SAR) (Reference 1) for the then-operating Savannah River Site (SRS) reactors contained only general descriptions of the plant design and procedural controls. Consisting of only a single volume, this document relied heavily on references as a substitute for the type of consolidated analysis that is typical of modern day SARs. Consequently, when design and procedure changes were proposed, it was necessary to review not only the body of the original SAR but also the cited reference documents. In some cases, even secondary references had to be reviewed as part of this change process. This was a cumbersome system that was prone to error and required numerous experienced personnel to review the information effectively. Revisions to the original SAR were made in December 1978, September 1983, and October 1987.

The process to upgrade the SAR so that it would fully describe the safety basis of the SRS reactors was begun by the Westinghouse Savannah River Company (WSRC) in early 1989. In the absence of specific DOE guidance, NRC Regulatory Guide (R.G.) 1.70, "Standard Format and Content of the Safety Analysis Reports for Nuclear Power Plants" (Reference 2) was selected as the primary reference document because of its applicability to commercial nuclear power plants. The upgraded SAR for the K-Reactor (Reference 3) was issued in July 1989 and approved for pre-restart activities. Separate SARs were also issued for the P- and L- Reactors. The subsequent Savannah River Special Projects Office (SRSPo) review of K-Reactor SAR was intensive (consuming over 20 man-months of effort) and resulted in over 100 pages of comments (References 4 through 7). Of the comments generated, slightly more than one-half required resolution prior to restart. The need for further upgrading was recognized, since some descriptions did not contain pertinent data that would define specific design information such as initial conditions, key nuclear-hydraulic parameters, safety-system setpoints, instrument errors, and response times. Also noted was the frequent use of "typical" or "sample" data when specific or appropriate bounding data were needed. (See Section 11.1 of this SER for the staff's evaluation of the Chapter 15 transients.)

In their SAR Upgrade Plan of February 23, 1990 (Reference 8), WSRC committed to further upgrade the K-Reactor SAR. The upgrade was to be accomplished in two stages. The first stage consisted of resolving the DOE comments that were within the operating safety envelope and incorporating them into amendments to be published prior to restart. During the second stage, the balance of the SRSPo comments would be incorporated into post-restart amendments.

5.1.2 Restart Criteria

WSRC shall submit for DOE review and approval a plan and schedule for upgrading the SAR in accordance with the criteria listed below. DOE will review the plan when received to determine which, if any, SAR sections are required to be upgraded before restart.

(1) Format and Content

The SAR format shall conform with the applicable requirements of NRC R.G. 1.70. The content of all safety-related sections shall conform with the depth requirements of the NRC Standard Review Plan, NUREG-0800 (Reference 9), in accordance with the DOE approved plan. Safety-related sections include all sections with the exception of Chapter 1, Introduction; Chapter 18, Human Factors; and Chapter 10, which was deemed not applicable to the K-Reactor and, therefore, not included.

(2) Accuracy

All safety-related systems and systems important to safety shall be accurately described in accordance with the DOE- approved upgrade plan. This description shall include an appendix listing safety-related systems and components that meet the safety-related equipment list requirements of Safety Evaluation Report, Section 5.6. All WSRC organizations that have a direct impact on safety or safety reviews shall also be described.

(3) Uses and Operations

WSRC shall require that all changes to plant operations be evaluated through the Unreviewed Safety Question (USQ) process to determine whether a change to the SAR is required. Administrative procedures shall be in place to ensure that the plant is operated in accordance with the SAR at all times. Procedures shall also be in place to restore the plant to SAR compliance and to fully report and analyze any operational incident outside the SAR. The requirements in this section must be developed and implemented before restart. In the interim, the Reactor Issues Management Program (RIMP) may be used to evaluate whether a change affects the SAR until the USQ process is formally implemented.

(4) Control and Update

The SAR shall be a controlled document. The procedures for controlling and updating the SAR must be submitted for DOE approval. These procedures shall be used to ensure that any proposed changes are thoroughly reviewed before being incorporated and that all changes are made expeditiously. A multidiscipline review of the proposed changes is necessary. All changes must (at minimum) be reviewed by the operating organization, the independent safety organization, and the SAR group within the laboratory. These changes shall also be submitted for DOE approval.

Records of all DOE approvals and conditions associated with the approvals shall be maintained in a retrievable manner for the life of the plant. Conditions on approvals shall be incorporated into WSRC procedures as appropriate. Each change stemming from a change to the plant (physical, procedural, or personnel) must be accompanied a USQ determination review. Where such changes are needed, the proposed SAR update should be approved by DOE before operating with the change in

place, unless it has been determined that there are no USQs.

(5) Accident Analysis

See the restart criteria for accident analysis, as discussed in SER Section 11.1, "Transient and Accident Analysis," for these requirements.

(6) Technical Specifications

A copy of the Technical Specifications to be used for each reactor at restart shall be upgraded periodically as changes occur and shall be included or referenced in Chapter 16 of the SAR.

5.1.3 Evaluation

WSRC provided the required K-Reactor SAR Upgrade Plan, which conformed to the SER restart criteria, on February 23, 1990. The WSRC SAR Upgrade Plan specified an update to the accident analysis; upgrades of the safety-significant chapters, consistent with the plant configuration; a description of the restart program; and updated administrative controls. The SAR restart criterion regarding NUREG-0800 level of detail is addressed by including a commitment to address this standard after restart. This commitment includes an additional comparison of the criteria, codes and standards, analysis, and programs required for a new commercial nuclear power plant with those used for SRS and a consolidation of information in the SAR beyond that required by other specific restart criteria. As explained below, the long-term SAR upgrade effort to address NUREG-0800 can be safely deferred until after restart. SRSP0 staff recommended approval of the WSRC SAR Upgrade Plan, and it was approved by the Director SRSP0 on April 17, 1990 (Reference 10).

SRSP0 and SRRO reviewed the upgraded K-Reactor SAR to determine if it complied with the approved upgrade plan. The chapters, which were rewritten in mid-1990, generally conformed with the plan; however, a number of additional new DOE comments were generated. For example, the SER requires a list of safety-related systems and components in the SAR, but the list has not yet been included or referenced. Other comments relate to the accuracy and usefulness of the SAR for making safety evaluations, establishing the Technical Specifications, and identifying the most currently applicable safety analysis bases or sources.

To avoid an open-ended review process, DOE and WSRC agreed that the following types of comments would be resolved and included in the upgraded SAR, which would be published as an amendment prior to restart:

- o comments that affect the definition or analysis of the safe operating envelope (or that are otherwise required by the SER)
- o significant new comments that pertain to the technical adequacy of accident analysis initial conditions, method of analysis, safety system setpoints, instrument errors and response times, and principal design criteria.

Comments that do not meet this criteria, but which require incorporation at a later date, would be documented in a change request in accordance with WSRC's document "Westinghouse Savannah River Company Procedures for Control of Nuclear Safety Documents, Procedure Manual - 11Q", August 1, 1990 (Reference 11). WSRC will maintain records of all comments that are not included in the amended pre-restart SAR and will include them in the long-term upgraded SAR, emphasizing the SAR's use as a reference for performing Unreviewed Safety Question Determinations (USQDs). (See SER Section 5.2.)

The above process for upgrading the K-Reactor SAR to modern day standards is generally equivalent to that of the commercial nuclear industry and is considered acceptable.

5.1.4 Conclusion

Progress has been satisfactory on the SAR upgrade program; however, the amended K-Reactor chapters have yet to be delivered and approved by DOE. SAR upgrades will remain an open item until WSRC has reviewed and upgraded the safety-significant chapters and resolved all applicable pre-restart comments as well as any other significant comments. DOE staff will evaluate the resolution of those comments to ensure that the SAR meets the restart criteria and will report on that evaluation in a supplement to the SER.

5.1.5 References

1. Safety Analysis of Savannah River Production Reactor Operations, DPSTSA-100-1, September 1972.
2. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," 1978 (as updated).
3. WSRC-RP-89-383, "Savannah River Site Reactor Safety Analysis Report, K Production Reactor," July 17, 1989.
4. Letter, R. E. Tiller to J. S. Moore, "Comments on the Interim Safety Analysis Report," August 7, 1989.
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8. Letter, J. L. Gallagher to R. E. Tiller, "SAR Upgrade Plan," February 23, 1990.
9. U. S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR

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11. Westinghouse Savannah River Company Procedures for Control of Nuclear Safety Documents, Procedure Manual - 11Q, August 1, 1990.

5.2 Unreviewed Safety Questions (USQs)

5.2.1 Background

The following paragraph defines an unreviewed safety question (as derived from References 1 and 2).

A proposed change, test or experiment or the discovery or identification of new information resulting from analyses of incidents or accidents, new or improved analyses or modifications or changes to procedures, processes, operations, structures, systems and/or components, shall be deemed to involve a USQ if (1) the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (2) the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (3) the margin of safety as defined in the basis for any technical specification is reduced.

Until recently SRS design changes were processed through a test authorization process that did not require a formal review to identify unreviewed safety questions (USQs). At SRS, conclusions on the acceptability of any new or added risk were based on a checklist of potential safety concerns related to design changes in conjunction with engineering or technical judgement. The USQ process requires specific, clearly documented reviews for Safety Analysis Report (SAR) and Technical Specification compliance and a detailed evaluation of risks if a design change is outside the operational envelope defined in the SAR or the Technical Specifications. At SRS, system conformance with the site's SAR and with technical specification assumptions and requirements was poorly documented, and specific reviews for conformance were not required in all cases.

The rigor and formality of the USQ process is intended to greatly reduce the probability of increased hazards being introduced through design changes. Although the NRC adopted a USQ process for use at commercial nuclear facilities, an equivalent process was not used at SRS. DOE Order 5480.6 (Reference 3) requires USQ evaluations; however, these evaluations were not fully implemented at SRS. There have been many significant design and operational changes at SRS; but, as of April 1989, none of these have been evaluated to identify USQs.

5.2.2 Restart Criteria

The criteria for determining the adequacy of the SRS USQ process for restart are described below.

- (1) A formal, controlled procedure shall be prepared for reviewing both proposed changes to the reactors and any new information to determine whether a USQ exists. The USQ procedure shall be reviewed and approved by DOE and implemented by the contractor prior to restart.
- (2) The USQ process shall be designed to evaluate new information or changes

to the plant to determine if they affect the existing plant design basis or operation in a significant manner. WSRC and DOE shall identify those individuals who are expected to use the USQ process and those individuals shall be trained in the process prior to restart.

- (3) The USQ procedure shall be based on the guidelines contained in NSAC-125, "Guidelines for 10 CFR 50.59 Evaluations"(Reference 4). WSRC shall submit USQ evaluations which conclude that a USQ exists to DOE for review and approval before operation or testing with that change in place. WSRC shall also submit to DOE an annual listing of those USQ evaluations for which it was concluded that USQs do not exist.
- (4) Procedures that are used when making a design change in the facility or change plant configuration and are in-place at restart shall reference the USQ determination procedure and require it to be used where applicable. The procedures shall also state the requirement for DOE approval before testing or operating with a change that involves a USQ. WSRC shall commit to using the USQ process for all post-restart design change activities.

5.2.3 Evaluation

The DOE staff reviewed the USQ determination procedure (Reference 5) for SRS and determined that it meets all the restart criteria except for the review of newly discovered information. The staff also determined that the procedure includes a requirement to review all proposed changes to the reactors, as required. The procedure does not consider new analyses and findings, which reveal previously unrecognized hazards to the safety of the reactors. This is an open item.

The procedure, which is based on the guidelines in NSAC-125, requires submittal to DOE of all USQ evaluations that conclude that a USQ exists. The procedure also requires all plant changes involving USQs to be approved by DOE before operation or testing with that change in place. In addition, the procedure contains a requirement to serialize USQ evaluations and submit to DOE all evaluations that conclude that no USQ exists.

Design and configuration control procedures have been implemented that require USQ screenings and evaluations to be performed before initiating changes.

USQ training is being provided to the proper personnel. The training is based on the guidelines of NSAC-125, as expanded below. The attendance lists demonstrate that operations, oversight, and management personnel have received training on USQs.

The course material for USQ training is based on NSAC-125 guidelines. The material defines the term "Unreviewed Safety Question"; requires an initial screening to determine whether the proposed activity could affect safety systems; then, requires a USQ Determination. The course explains how to answer and document the answers for the seven questions that NSAC 125 (and 10 CFR 50.59) require for USQ determinations. WSRC has a list of qualified USQ reviewers who have received more extensive training and are experienced in USQ

reviews. WSRC has committed to annual retraining for these reviewers.

Resources adequate to implement the procedures in a timely manner have been allocated.

5.2.4 Conclusion

Based on the above evaluation, the DOE staff concludes that WSRC has developed a procedure for determining whether a USQ exists. This procedure does not address newly-discovered information. This is an open item.

The procedure requires DOE approval for activities that involve a USQ and a submittal of those evaluations that conclude that no USQ exists. Design and configuration control procedures also reference the USQ procedure.

5.2.5 References

1. Title 10, Code of Federal Regulations (CFR) Part 50.59.
2. Title 10 CFR, Part 830.59, Draft, July 26, 1990.
3. U. S. Department of Energy, DOE Order 5480.6, dated September 23, 1986.
4. NSAC-125, "Guidelines for 10 CFR 50.59 Evaluations," dated June 1989.
5. WSRC, Reactor Restart Division Administrative Manual, RD-1, RDP-12.03, Rev. 0, "Unreviewed Safety Question Initial Screening and Evaluation Procedure," October 26, 1990.

5.3 Technical Baseline Program

5.3.1 Background

NRC Regulatory Guide (RG) 1.64 Revision 2, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (Reference 1) states that with certain modifications American Nuclear Standards Institute (ANSI) Standard N45.2.11-1974 (Reference 2) is acceptable for providing an adequate basis for complying with the quality assurance requirements of Title 10 of the Code of Federal Regulations (CFR) Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," (Reference 3). ANSI N45.2.11-1974 stipulates requirements and recommendations for establishing and executing a quality assurance program during the design phase of nuclear power plants. Many commercial industry nuclear plants that were designed and constructed prior to the established guidelines set forth in RG 1.64, Revision 2, were not fully in compliance with 10 CFR 50, Appendix B. Design baseline reconstitution programs have been implemented in an effort to recover plant design bases for these plants.

The SRS reactors are similar to those at older vintage commercial plants in that they were designed and constructed without a well-defined and established quality assurance program. In 1988, under DOE direction, WSRC committed to establishing and implementing a Technical Baseline Program (TBP) at SRS. The impetus for that direction and commitment came from the development of a Reactor Safety Improvement Program (RSIP). One objective of the RSIP was to establish the Technical Baseline for SRS reactors by confirming that systems and components important to safety are capable of performing their design functions when called upon.

In 1988, a steering committee made up of prime contractor and DOE personnel was established to develop a charter and workscope for the Technical Baseline Program. The steering committee was also responsible for monitoring the performance of the Technical Baseline work according to the approved task plans. According to the management plan (Reference 4), the immediate objective of the technical baseline program was to demonstrate and verify the capability of selected reactor systems to provide safety functions in response to the more significant design basis events (DBEs) in the Safety Analysis Report (SAR). The initial effort was directed towards identifying important safety functions and system and component design and performance requirements and comparing the system and component capabilities to those requirements.

The initial effort for the technical baseline program began in L-Reactor, because it was shutdown and was accessible for performing walkdowns (K and P reactors were operating). Technical Baseline work was shifted from L-Reactor to K-Reactor when the K-Reactor restart effort began.

5.3.2 Restart Criteria

The criteria for determining the adequacy of the SRS technical baseline program for restart are described below.

- (1) Baseline walkdown inspections shall be performed to establish (on the basis of a representative sample of safety systems) that the as-built conditions correspond to the safety analysis. Safety-significant discrepancies identified by the TBP shall be evaluated and resolved prior to restart by the Restart Issues Management Program (RIMP) Committee.
- (2) A determination of the number of additional baseline walkdowns required before restart shall be made, and the safety rationale for that determination shall be documented.
- (3) Material condition walkdowns shall be performed on the following: (1) Critical Monitoring System, (2) Supplementary Safety System, (3) Airborne Activity Confinement System, (4) Cooling Water System, and (5) Process Water System. Safety Significant discrepancies identified by the walkdowns are to be evaluated by the RIMP Committee and resolved prior to restart.
- (4) A determination of the number of additional material condition walkdowns required before restart shall be made, and the safety rationale for that determination shall be documented. This is to be performed in conjunction with the performance of satisfying Restart Criterion 2, above. The results of the determinations and their bases shall be submitted to DOE for review on a schedule to be agreed to by DOE. The schedule shall facilitate DOE review of the results before restart.
- (5) The TBP shall establish system boundaries and interfaces on all safety systems identified for restart baselining activities.
- (6) The TBP shall identify all existing documentation needed to support the design basis of the systems chosen for restart. If the design documentation is unavailable or its accuracy is uncertain, a plan of reverification of the missing information shall be developed and its implementation shall be tracked. Justification shall be given as to the determination of its safety significance and, if it is deemed to be a technical issue, an additional determination of its restart significance shall be made according to the RIMP assessment criteria.
- (7) The TBP shall comply with its approved task plans and quality assurance requirements throughout the restart effort.
- (8) Provision shall be made for the TBP documents to be updated and controlled as described in SER acceptance criteria 5.4 and 5.5.
- (9) WSRC shall establish and implement (on a schedule to be approved by DOE) a configuration control system to protect the integrity of the documentation and information established in the TBP effort, as described in SER acceptance criteria 5.4 and 5.5.

The configuration control program discussed in item 9 will include some long-term requirements. WSRC shall meet these requirements post-restart.

5.3.3 Evaluation

Restart Criterion 1

An evaluation of the Technical Baseline Program in the areas of baseline walkdown inspections, material condition walkdowns, system boundaries and interface, design basis documentation, compliance with quality assurance requirements, TBP document control was conducted.

In evaluation the adequacy of baseline walkdown inspections, the following WSRC procedures were reviewed: SRTB-88-011 (Reference 5), RTR-2597 (Reference 6), SRTB-88-019 (Reference 7), SRTB-88-020 (Reference 8), SRTB-88-040 (Reference 9), SRTB-88-037 (Reference 10), SRTB-88-63 (Reference 11), and SRTB-88-067 (Reference 12). These procedures contain the methodology to adequately:

1. identify the safety related system functions as defined in the Safety Analysis Report (SAR);
2. select the representative sample of safety systems to be used as the basis for establishing that the as-built conditions correspond to the safety analysis;
3. identify the components essential in performing the system's safety function;
4. document the as-built configuration of the selected systems and components;
5. identify the system and component functional requirements and capabilities necessary to perform the identified safety functions;
6. compare the requirements for systems and components with their capabilities in order to determine the adequacy of the current as-built configuration; and
7. identify discrepancies and ensure their resolution in a timely manner.

RTR-2597, "Task 2 System Selection," provides an adequate basis for selecting a representative sample of safety systems. The procedure considers the important features of system operability for each design basis event (DBE) and provides flexibility for other considerations, such as the nature of the systems (fluid versus electrical) and the level of industry experience with similar systems. The methodology used for system selection evaluates the role (active or passive), actuation mode (automatic or manual), environment (harsh or mild), and function (primary or secondary) of each system. Additional factors are applied to account for the number of safety functions the system satisfies, critical timing requirements, reliability characteristics, the history of operation, and modifications to the system.

Implementation of RTR-2597 resulted in a total of six safety systems being selected as the representative sample for baseline walkdown inspections.

These systems were water removal system; process room spray system; emergency cooling system; discharge assembly cooling system; scram system; and process water diesel generators.

A baseline walkdown inspection was conducted by WSRC on each of the six safety systems during the time period of November 1988 to June 1989. WSRC engineers conducting the inspections were instructed to follow the requirements in SRTB-88-020, "Savannah River Site Reactor As-Built Walkdown." This procedure required various component data to be documented on a component data sheet. The data included component type primary "as-built" and "as-designed" source, location, and nameplate information. This data in conjunction with system requirements and capability determination required by SRTB-88-040, SRTB-88-037, and SRTB-88-063 provide adequate assurance that all information available to support the safety analysis was obtained. Any missing information was required to be documented on discrepancy data sheets for technical resolution. SRTB-88-020 requires a formal disposition process for all discrepancies. The process classified each discrepancy as requiring either "no change," "issuance of an NCR," "referral to the Plant Manager for action as appropriate," or "issuance of a drafting request to update the drawing." SRTB-88-020 also requires that baseline walkdown inspection discrepancies be documented. In addition, discrepancies and inconsistencies between the as-designed plant configuration and the as-built configuration identified during inspection were required to be brought to the attention of the Technical Baseline Program Director. If the reported discrepancy was determined to represent a difference between design capability and the safety analysis, the condition was declared a technical issue and presented to the Restart Issues Management Program (RIMP) for resolution. A total of 37 technical issues were identified by the baseline walkdown inspections, all of which were evaluated and dispositioned by the RIMP committee in accordance with the requirement in SRTB-88-67.

In conclusion, a representative sample of safety systems was inspected to establish the as-built conditions in the reactor plant. This information was then used as an input during the SAR upgrade. Also, all safety-significant discrepancies identified by the TBP have been evaluated and resolved by the RIMP committee.

Review of applicable procedures, field observations on a sampling basis, and review of inspection results, including disposition of identified discrepancies, concluded that the information documented in the TBP adequately reflects the as-built system conditions and was therefore adequate to incorporate into the SAR upgrade.

Restart Criterion 2

Documentation of a determination of the number of additional baseline walkdowns required for restart and the safety rationale for the determination is a restart requirement. WSRC has completed this determination and documented the results, including the safety rationale, in Technical Position Paper TPP-002 (Reference 13). TPP-002 provides an overview of the process used to establish the adequacy of the six systems selected as a representative sample of safety-systems. TPP-002 also presents a summary of the results of

the baseline walkdown inspections, a listing of the identified technical issues requiring resolution prior to restart, and discusses the adequacy of the sample of safety systems inspected when considered in conjunction with other upgrade/improvement initiatives underway in support of reactor restart.

TPP-002 discusses the relationship of the TBP to other restart programs such as the restart test program, the seismic program, the PRA, the Technical Specification upgrade, the Safety Analysis Report (SAR) rewrite, the configuration management program, and the environmental qualification program. TPP-002 concludes that these programmatic initiatives interact in order to provide reasonable assurance that significant safety issues are identified and resolved and that the plant corresponds to the SAR; therefore, no additional baseline walkdown inspections are required for restart.

Review of TPP-002 and the applicable safety rationale has concluded that WSRC has performed an adequate determination of whether any additional baseline walkdown inspections are required, and that the conclusion that no additional baseline walkdown inspections are required before restart is appropriate.

Restart Criterion 3

SRTB-89-281 (Reference 14) provides adequate procedural guidance for conducting material condition walkdowns of selected systems. The selected systems specified in SRTB-89-281 are: (1) Critical Monitoring System, (2) Supplementary Safety System, (3) Airborne Activity Confinement System, (4) Cooling Water System, and (5) Process Water System.

The inspection requirements specified in SRTB-89-281 are sufficient to detect visible component damage or degradation which may impact the component's ability to perform its designed safety functions in the prevention or mitigation of DBEs as discussed in the SAR.

WSRC has completed the required material condition walkdowns, and the safety significant discrepancies identified have been evaluated and dispositioned by the RIMP committee.

Restart Criterion 4

WSRC has completed the required evaluation/determination of the need for additional material condition walkdowns and the safety rationale behind that determination. The evaluation is contained in WSRC Technical Position Paper TPP-006, "A report on the Walkdown Findings."

The Technical Position Paper states that "many of the findings relate to attributes that do not affect component or system operability." Sixty discrepancies relate to general cleanliness and housekeeping; 124 relate to the tagging or labelling of components; 69 relate to personnel safety covers. This accounts for 256 of 766 or one-third of the material condition discrepancies identified. The remaining two-thirds consists of various component material condition discrepancies which were determined not to

immediately impact component operability. TPP-006 also discusses the fact that the system's ability to perform its safety related functions will be further confirmed during testing.

WSRC has concluded that the number and type of findings is not indicative of a widespread problem that could adversely affect reactor safety. This conclusion is documented in OPS-RRP-900137 (Reference 15). The WSRC analysis of the findings indicates that most material conditions discrepancies are due to inadequate maintenance and poor work practices. These root causes are being addressed on a plant-wide basis under the SER sections on maintenance and housekeeping (13.0 and 10.5, respectively).

One technical issue was identified. The issue deals with the use of Teflon tape in selected AACS instrument Air Piping. The material can break down and cause improper operation of instrumentation due to leaks or plugging of instrument air orifices, if introduced into instrument air lines. WSRC RIMP determined that it does not represent a significant risk to the health and safety of the public because:

1. Complete loss of instrument air pressure would not degrade safety function integrity since dampers and latches fail in the desired position, and
2. There is a backup source of compressed air from the Standby Air System.

The determination fails to consider the impact of pieces of degraded Teflon tape in the air system migrating to instrumentation and causing binding between moving parts such as solenoid valve seats, plugs, and stems. No amount of backup air supplies will mitigate this failure. Where component movement is required to achieve its fail safe position on loss of air, achieving the proper fail position is dependent on free, unbound movement of component parts. Pending adequate resolution of this technical issue, compliance with this restart criterion is an open item.

In summary, given the extent of walkdowns performed, the nature of discrepancies identified and the other plant improvements being implemented prior to restart, as identified in the WSRC position paper and the above evaluation, it is concluded that no additional material condition walkdowns are required. However, the item will remain open pending adequate resolution of the Teflon tape technical issue.

Restart Criterion 5

Evaluations of TBP procedures SRTB-88-020, "Task 5, As-built Walkdowns," Rev. 0, and SRTB-89-281, "Material Condition Walkdowns," were performed. Although the procedures appear to describe a satisfactory program to establish system boundaries and interfaces on all safety systems identified for restart baselining activities, the staff's review of system walkdown boundaries determined that not all critical components were included within the boundaries. Pending revision of the applicable documents to include all critical components this is an open item.

Restart Criterion 6

The Technical Baseline Documents (TBDs) generated for six systems document comparisons between requirements and capabilities for system/component functional design attributes. For some design attributes, information was not available regarding either the requirement or the capability. The missing information was considered to be a technical issue if it could significantly affect the ability of the system to mitigate the consequences of DBE. Other missing information was documented in another section of the TBD under the category of Candidate Verification Items (CVIs). Technical Issues were processed through the Reactor Issues Management Program (RIMP) and were either resolved for restart or resolution was deferred to the long-term Reactor Safety Improvement Program (RSIP).

RRD-TBP-090-0594 (Reference 16) describes the tracking system implemented for closing out CVIs. The document identifies the various fields in the database used to describe the CVI; identify its current ranking status and identify the person responsible for its resolution. The document assigns the Technical Baseline Support Lead Engineer with the responsibilities of ensuring implementation and maintenance of the CVI tracking program and of issuing passwords to prospective users.

In summary, based on the discussion above, the Technical Baseline Documents (TBD) identify all existing documentation needed to support the design basis of the systems chosen for restart. If the design documentation was unavailable or its accuracy uncertain, the method of reverification of the missing information was the generation of a Candidate Verification Item (CVI) and tracking of the item through to resolution. Justification was given regarding the safety significance of the item and, if it was deemed to be safety significant and a technical issue, the item was forwarded to the RIMP committee and an additional determination of its restart significance was made according to the RIMP assessment criteria. Consequently, the requirements of Restart Criterion 6 have been satisfied.

Restart Criterion 7

To verify program compliance with the approved task plans and quality assurance requirements throughout the restart effort, DOE reviewed RTR-2600 (Reference 17). The review found three chapters with requirements relevant to program compliance:

Chapter 2, Paragraph 2.1, of RTR-2600 states: "This Quality Assurance Plan is applicable to all organizations responsible for activities required for the Technical Baseline Program."

Chapter 5, Paragraph 5.1.1, of the QA Plan specifies that "Personnel assigned to the Technical Baseline Program shall perform activities in accordance with the latest revision of applicable approved and controlled instructions and/or procedures."

Chapter 18 of RTR-2600 provides the following direction: "Before the completion of the Technical Baseline Program, the Program Director will request a surveillance of the entire program in accordance with the Savannah River Plan Quality Assurance Manual QAP 18.3"

The three requirements jointly direct Technical Baseline Personnel (TBP) to comply with applicable procedures/instructions and provide for verification of compliance by the Reactor Quality Department.

In summary, the required surveillance, if performed in sufficient depth, will provide adequate assurance of compliance with the approved program task plans and quality assurance requirements; however, until this surveillance has been completed and reviewed for adequacy this is an open item.

Restart Criterion 8

WSRC is required to assure that TBP documents are properly updated and controlled. RP5.1303 (Reference 18) was reviewed and determined to contain sufficient information to enable users to implement design controls to update Technical Baseline documents in a manner consistent with the site QA requirements for restart.

The procedure assigns authority for the Technical Baseline Documents to Reactor Engineering, requires changes be initiated, evaluated, reviewed, approved, and incorporated as needed. The procedure further requires a full review and update as needed every three years.

In summary the procedure contains provisions for the TBP documents to be updated and controlled. For further discussion of document control, see SER section 5.5.

Restart Criterion 9

The requirement for WSRC to establish and implement a configuration control system to protect the integrity of the documentation established in the TBP is provided in this SER section for continuity and is addressed in SER section 5.4.

5.3.4 Conclusion

Based on the above evaluation, the DOE staff concludes that WSRC has made acceptable progress in the technical baseline program. Open issues are clearly identified and their resolution will be documented in a supplement to the SER.

5.3.5 References

1. Regulatory Guide 1.64, Revision 2, "Quality Assurance Requirements for the Design of Nuclear Power Plants," U. S. Nuclear Regulatory Commission, Office of Standards Development, June 1976.

2. ANSI/ASME N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants."
3. Appendix B of Title 10 of the Code of Federal Regulations (CFR) Part 50, "Quality Assurance Criteria For Nuclear Power Plants and Fuel Reprocessing Plants."
4. RTR-2586, "Reactor Technical Baseline Program Management Plan," J. L. Hendrix, WSRC, Director, Technical Baseline Document Program, September 22, 1988.
5. SRTB-88-011, "Task 2 - Shutdown Logic Diagram (SLD) Development," S. R. Whitsett, Impell Corporation, October 26, 1988.
6. RTR-2597, "Task 2 - System Selection," Enno Nomm, December 20, 1988.
7. SRTB-88-019, "Task 3 - Safety Function Diagram Development," S. R. Whitsett, Impell Corp., Revision 1, May 15, 1989.
8. SRTB-88-020, "Task 5 - As-Built Walkdowns," L. A. Wild, Combustion Engineering, Revision 1, April 12, 1989.
9. SRTB-88-040, "Task 6 - Reactor System and Component Functional Requirements Development Plan," T. R. Enright, Impell Corporation, October 28, 1988.
10. SRTB-88-037, "Task 7 - Development Plan for Establishing Functional Capabilities," D. R. Wade, Combustion Engineering, November 1, 1988.
11. SRTB-88-63, "Task 8 - Identification of Critical Component Design Capabilities," D. R. Wade, November 1, 1988.
12. SRTB-88-067, "Task 9 - System Evaluation and Development of Draft Technical Baseline Document," T. G. Clark, Revision 1, June 15, 1989.
13. OPS-RRP-900024 Memorandum from P. W. Dickson to J. L. Gallagher, Technical Position Paper: Technical Baseline Program, February 27, 1990.
14. SRTB-89-281, "Material Condition Walkdowns," L. A. Wild, April 12, 1989.
15. OPS-RRP-900137, "Technical Position Paper: Material Condition," August 28, 1990, P. W. Dickson to J. L. Gallagher.
16. RRD-TBP-090-0594, "Candidate Verification Item Tracking System," October 18, 1990, R. M. Gregory to Fred Beranek.
17. RTR-2600, "Reactor Technical Baseline Program Quality Assurance Plan," J. L. Hendrix, October 17, 1988.
18. RP 5.1303, Revision 0, "Reactor Engineering Design Authority Document Updates," March 12, 1991, effective April 15, 1991.

5.4 Design Change Process

5.4.1 Background

DOE Orders and WSRC procedures document the need for adequate design control. The WSRC Reactor Restart Division (RRD) and Engineering & Projects Division (E&PD) are in the process of implementing these design change control requirements at the working level. Having recognized the need for improved design change control practices, WSRC is working toward a more consistent and complete design change process. The lack of a formal, documented and controlled design change control program in the past contributed to a general inability to match engineering documents to the safe operating envelope of the reactors. In addition, a complete set of controlled and verified as-built engineering and operations documentation is not available for the SRS reactors, design and performance materials are not always traceable through controlled documents, and there is a lack of as-built system designs that clearly indicate the full-spectrum of design and performance requirements.

5.4.2 Restart Criteria

The criteria for determining the adequacy of the SRS design change process for restart are described below.

- (1) WSRC shall establish organizational structures, and identify line responsibilities and interfaces to ensure compliance with the change control and configuration management requirements for reactor restart.
- (2) WSRC shall ensure that the design change control process meets the criteria of the 1988 Savannah River Plant K Reactor Restart Strategy that address technical baseline efforts, configuration management, design changes, and drawing and document control.
- (3) Rigorous design change control and configuration management programs shall be required for all systems within the Technical Baseline Program. For the remaining systems, WSRC shall implement design change control and configuration management programs for:
 - o any new system or changes being implemented for restart
 - o associated system-plant interfaces (sufficient to ensure design adequacy)
- (4) Reactor restart program design changes and the associated interfaces to safety and non-safety systems, structures and components shall receive sufficient review to ensure there are no potential adverse effects to as-built safe shutdown systems. Design changes to safe shutdown systems, structures, and components are to receive both independent and multi-discipline reviews. The review process shall have sufficient depth and breadth to ensure design and performance adequacy. Reviews should address:

- o design basis and functional requirements
- o system design and performance
- o system interface and interaction
- o equipment environmental and seismic qualifications
- o technical specifications
- o other safety analyses and reviews as required by applicable DOE orders

Changes should also be reviewed through the Unreviewed Safety Question process.

- (5) Personnel performing formal design review shall have the necessary technical competence as evidenced by education, training, and experience.
- (6) The WSRC quality assurance program shall include audits to monitor and appraise compliance with the design change control and configuration management program requirements for restart. These program requirements shall be to a level commensurate with those set forth in the governing design input requirements for the system.
- (7) A combination of selected, formal design reviews and in-depth limited scope reviews shall be performed to assess the adequacy of the design, installation and testing activities for K-reactor. These reviews should be used in combination with the startup test program to provide reasonable assurance of reactor facility functional and design adequacy for restart.
- (8) The program plan shall establish design basis, design control, and configuration management program upgrades to be implemented post - restart. That program shall include a requirement that system boundaries and interfaces be defined for all safety related systems and all design changes be reviewed to ensure that no unacceptable systems interactions could occur as a result of the proposed changes. The program plan and implementation schedule shall be submitted to DOE for review and approval.

5.4.4 Evaluation

Restart Criterion 1

WSRC has proposed and developed specific Administrative Instructions (AIs) that define and delineate specific lines of responsibility and review levels required for design changes and reviews. The primary definition of the organizational structure being used by WSRC is set forth in Technical Specification Chapter 5, "SRS Administration Controls", Rev. 0, issued December 18, 1990 (Reference 1). In this Technical Specification, Chapter

5.3, WSRC defines the specific roles and responsibilities of the Reactor Restart Division (RRD), the Savannah River Laboratory (SRL), the Engineering and Project Division (E&PD), the Environmental, Safety, Health and Quality Assurance Division (ESH&QA), and the Human Resource section.

The RRD is broken into nine subsections by Chapter 5.3.1 of the Technical Specifications. These subsections are as follows:

- o Reactor Operations
- o Reactor Startup and Testing
- o Reactor Engineering
- o Planning/Scheduling and Cost Control
- o Reactor Training and Procedures
- o Technical Department
- o Reactor Quality Assurance and Assessments
- o Reactor Strategic Programs
- o Configuration Management Department

The Operations Department is assigned the line responsibility for facilitating physical changes to the plant configuration and coordinates the activities of all work groups in the reactor area, thereby assigning them the task of specifically being cognizant of and in control of the plant's physical configuration.

The Reactor Engineering Department is assigned the responsibility of providing a liaison function between Reactor Projects and the Configuration Management Departments. The Reactor Engineering Department is also tasked to be the Design Authority for the project which makes the Reactor Engineering Department the line organization responsible for the plant's design configuration.

The creation of the Configuration Management Department further indicates the commitment of WSRC to the overall support of Configuration Management required by this restart criteria. It is the charter of this department to develop both long and short term configuration management controls.

The overall effect of Technical Specification, Section 5.3 is to define precisely the duties and responsibilities as related to configuration control for the restart of the SRS reactors.

Chapter 5.3.3 is of particular importance to this SER topic, as it defines the organization of the E&PD. By the inclusion of this organization in this document Bechtel Savannah River Inc. is captured under the "umbrella" of the WSRC Quality Assurance Manual (Reference 2) which has provisions for auditing

compliance with the established Configuration Management Program.

Chapter 5.3.4 deals with the departmental charter of the ESH&QA. Within this chapter specific responsibilities are assigned to the four subsections of this department. Paragraph 5.3.4.1 defines the QA department and assigns to it the overall responsibility of surveillance and oversight of the SRS Quality Assurance Manual. Within this manual are specific procedures that control configuration, which will be discussed later in this evaluation. Paragraph 5.3.4.2 defines the Safety Department and provides it with the task of monitoring the operation and startup of the SRS reactors and ensuring that DOE Orders are followed. It also provides independent oversight of operations to ensure compliance with procedures in place and assists in the development of new procedures.

Precise controls are established by Chapter 5.5 of the Technical Specifications. Paragraph 5.5.1 deals with the contractual agreement DE-AC-09-88-SR18035 (Reference 3) between Westinghouse and DOE. This paragraph specifically cites the Technical Specifications as the primary source of control. Paragraph 5.5.3 deals with definition of Technical Specifications and, in particular, defines section 5 of the Technical Specifications as being dedicated to the development of and compliance with Administrative Controls. These specific means of control will be developed later in this evaluation in the form of work and configuration control procedures.

In the remaining seventeen paragraphs of Chapter 5.5, further controls for reactor operations and startup are provided. Several paragraphs, while appearing to have no bearing on configuration control, will require configuration control as an initial condition and will provide further means of controlling configuration as defined by "Technical Specifications" (par. 5.5.3), "Charge and Discharge Operations" (par. 5.5.4), "Unreviewed Safety Question Determination Procedure" (par. 5.5.5), "Reactor Startup Authorization" (par. 5.5.6), "Test Authorizations" (par. 5.5.7), "Maintenance Operating, Abnormal Condition Control, Special and Administrative Procedures" (par. 5.5.8), "Process Technical Manuals" (par. 5.5.9), "Qualification and Training Requirements" (par. 5.5.10), "Emergency Plan - Reactor Plan" (par. 5.5.11), "Emergency Plan - Site Plans" (par. 5.5.12), "Security Plan" (par. 5.5.13), "Fire Control Pre-Plan" (par. 5.5.14), "Reactor Fire Control Plan" (par. 5.5.15), "Core Operation Report" (par. 5.5.16), "Core Shutdown Report" (par. 5.5.17).

All of the methods discussed above will provide specific means to control configuration. By putting in place Technical Specification Chapter 5, SRS Administrative Controls, Rev. 0, December 18, 1990, and the other work control procedures that will be discussed later in this evaluation, WSRC has met the requirements of Restart Criterion 1.

Restart Criteria 2 and 3

WSRC has completed specific tasks which will satisfy Restart Criterion 2. These are:

- (1) The establishment of a Technical Baseline (TB) program, which has been completed on six DOE-designated, safety-related systems for K reactor.
- (2) The scheduled completion of the TB program as part of a long-term program after restart.

The completion of these task satisfies the "Technical Baseline" portion of Restart Criterion 2. More specific requirements and evaluation of the Technical Baseline Program are discussed in Section 5.3 of this report.

Restart Criterion 3 requires rigorous design change control and configuration management programs to be instituted for (1) all systems within the Technical Baseline Program, (2) any new system or change being implemented for restart, and (3) the associated system-plant interfaces sufficient to ensure design adequacy. WSRC has implemented several procedures. These are:

- (1) Unreviewed Safety Question Initial Screening and Safety Evaluation Procedure RDP 12.03, Section 12.03, Rev. 0, October 26, 1990 (Reference 4)
- (2) Design Authority Review, Reactor Technology Department Standard Practice Manual, Cat. 3, Sec. 8.1, Rev. A (Reference 5)
- (3) Reactor Restart Division Work Control Procedure DMP-3.01, Rev. 1 (Reference 6)
- (4) "Reactor Plant Modification and Design Control Program", RDP-5.01, Rev. 0 (Reference 7)
- (5) "Technical Review of Work Packages," RP-5.1601, Rev. 0 (Reference 8)

By implementing RDP 12.03, RP-5.1601, and RDP 5.01, WSRC has committed to evaluate all activities that may change the configuration of the facilities. As a part of the Unreviewed Safety Question (USQ) review, the proposed activity must be evaluated to the criteria set forth in the Technical Specifications and the Safety Analysis Report (SAR). Seismic concerns and designs are both covered by the Technical Specifications and the SAR and are addressed by the USQ process. The procedure further requires that the USQ evaluation be approved by DOE prior to implementation of the activity when a USQ is determined to exist. The requirement to use the USQ process applies to all systems thereby enveloping the requirements of restart criterion 3.

Paragraph 5.3.3 of the procedure requires that seismic considerations be specifically applied during the USQ review. Paragraphs 3.6, 3.7, 3.8, and 3.9 establish, by definition, the requirements for evaluators, implementors, and qualified reviewers. These paragraphs also establish training requirements and minimum qualifications for these individuals. The Design Review Authority (DRA) procedure establishes the review process for Reactor Engineering and defines a controlled tracking system for all design changes. The procedure defines a strict design review and ensures a multi-tiered review. The design

changes must be reviewed by the system engineer, the configuration management administrative group, and reactor operations engineering. The procedure also requires the change to be reviewed in accordance with RDP 12.03.

DMP-3.01 ("Work Control") requires that all modifications be controlled in accordance with a rigidly defined system of administrative and scheduling considerations. Once an activity is identified, evaluated and scheduled by the previously discussed procedures, the actual implementation will be governed by the controls described by this procedure. Features include a system planner, who schedules with operations and the implementing organization to perform the activity with a minimum impact to facility activities and ensures that the surveillances necessary to verify that the requirements of Restart Criteria 2 and 3 have been completed. The above discussed procedures together with a rigorous training program on these procedures meet the requirements of restart criteria 2 and 3 to control activities and configuration on plant systems.

DPSOLs 105-1933 and -1934 (References 9 and 10) require that following modifications of seismically qualified systems, walkdowns will be performed by qualified engineers. DPSOL 1934 requires the operations/maintenance department notify seismic engineering of their intent to modify or work on a seismically qualified system or component. DPSOL 1933 provides the administrative path by which Engineering documents a walkdown to assure that the seismic capabilities of the modified system have not been compromised. The firm requirement that no work be done outside of the established administrative avenues reinforced by training in these areas as discussed above provide the assurances required by restart criteria 2 and 3.

Drawing and document control is discussed in Section 5.5 of this report.

Restart Criterion 4

The procedures discussed above satisfy Charlotte Criterion MS-21.1 which calls for the implementation of a formal design change process. They also satisfy Charlotte Criterion DB-2.11 in that they establish a method by which modifications to seismically qualified systems will be reviewed for impact, control, and verification. These procedures also require the establishment and control of specific documentation and require design control for Technical Baseline systems as well as the balance of systems in the facility. Changes are to be reviewed in accordance with RDP 12.03, "Unreviewed Safety Question Initial Screening and Safety Evaluation Procedure".

RDP 5.01 is a very complex procedure and touches many organizations with the requirements that it imposes. Paragraphs 4.1.7.2, 3.1.C, 3.1.D, and 3.2.A specifically require that both Reactor Engineering and E&PD review and assure that the design does not violate the design basis (as known) or that the functional performance requirements of the design will meet the intention and requirements of the facility. Paragraph 3.1.N also requires that the post modification testing be reviewed to assure that the test data confirms the design. This paragraph, further, provides the data required for updating: (1) the system's design and performance (paragraph 4.1.7.3), (2) the operational

procedures required to be revised after modification (paragraph 4.1.7.3 and 3.1.H), and training requirements (paragraph 3.1.H and 3.5), all required as a result of the implementation of this procedure.

System interface and interaction is reviewed as a part of the review process associated with the implementation of the USQ procedure and also by paragraph 3.1.G which requires that the design be reviewed by Reactor Engineering to assess the design's impact on the Technical Specifications and the Safety Analysis Report.

Procedure DMP 3.01 delineates the requirements for work control. In paragraph 3.14-A.4, Reactor Engineering is specifically tasked with the review of work packages to assure their compliance with special program requirements such as seismic and environmental.

RDP 5.01 has a provision for an independent multi-disciplined review board chaired by the Reactor Engineering Department. It is the task of this board (per paragraph 4.1.7.5) to provide this review. This task applies to "Project" level Modifications. Lower level "Operating Expense" projects are handled by the requirements of DMP 3.01. These Design Changes/Modifications are of a much smaller scope than those handled by RDP 5.01 (less than \$50K). In DMP 3.01 there are specific requirements for review by the Maintenance section (a discipline orientated organization), Reactor Engineering, and Operations. Although the review process is in place, the lack of specific provision for a "independent multi-discipline" requires that this item remains open until such provision can be made.

Restart Criterion 5

WSRC has implemented "Section Level Instruction System Cognizant Engineers Desk Top Instruction, Category 4" (Desk Top Instructions), Rev. 0 (Reference 11). In this procedure, it is clearly detailed that the cognizant system engineer initiates or, at a minimum, participates in the design and implementation of design changes and modifications for the systems assigned to him. Chapter 5.3 of this procedure further tasks the cognizant engineer with specific design review duties and responsibilities. Chapter 5.1 of the procedure specifically outlines qualifications for cognizant system engineers. Comparing the requirements set forth in Restart Criterion 5 of this evaluation with the requirements set forth in the procedure being discussed in this paragraph, it is clear that the intent of Restart Criterion 5 will be met by the implementation of this procedure. In addition, Institute of Nuclear Power Operations (INPO) Guideline 85-031, "Guidelines for the Conduct of Technical Support Activities at Nuclear Power Facilities" (Reference 12), and INPO Good Practices TS-413, 85-033, "Use of System Engineers" (Reference 13), were used in the preparation of the procedure, which makes it comparable to those procedures used in commercial nuclear facilities.

This issue is considered an open item pending issuance of the Desk Top Instructions or its equivalent as a final document.

Restart Criterion 6

Chapter 5 of Rev. 0 of the Technical Specifications, paragraph 5.3.4.1, establishes requirements for the Quality Assurance Department. It is the charter of this Department to develop a Quality Assurance (QA) Manual and a program for audits and the reporting of findings. Chapter 3 of the WSRC QA manual is devoted solely to Design Change Control. Chapter 3 is divided into three sub-chapters (3.1, 3.2, and 3.3) that deal with Category I Projects (cost > 1.0 Million); Category II Projects (cost ≤ 1.0 Million and construction/installation cost ≥ 50K); and Projects with installation cost ≤ 50K. These sections deal specifically with the development of inspections plans and requirements and the overall establishment of verification of engineering specifications.

The designation of design classifications is established by Chapter 2 in the QA manual. Specifically, sub-chapter 2-1 defines the classifications (listed in descending order of importance) as:

- o nuclear safety
- o critical protection
- o production support
- o general services

It is specifically stated in this section that greater scrutiny will be applied to those with higher classifications; however, all classifications will be subject to basic QA surveillance and control. The lower classifications were designated only to allow for a more selective application of QA. Chapter 18 of the QA manual establishes a means to audit and perform surveillances on the effectiveness of compliance with the QA manual and its programs. Sub-chapter 18-1 specifically delegates the responsibility to the Manager of the Quality Assurance Department to establish a program and schedule for these audits. In accordance with subchapter 18-1, the manager of Quality Assurance Audits shall establish an annual audit schedule and as a minimum design, construction operations, and research and development shall be audited annually.

In sub-chapter 18-2, the scope of audits is defined to be applicable to all research, development, design, procurement, construction, fabrication, operation, and decommissioning of the SRS reactors. It further establishes the commitment to report findings as either nonconformances or corrective action reports for tracking and ultimate resolution. Sub-chapter 18-3 sets forth a plan to audit external organizations for their compliance with the QA manual. This allows for the audits of vendors and suppliers and further demonstrates an overall commitment to the assurance of Quality in all aspects of design and its control. The items discussed above demonstrate that the requirements set forth in Restart Criterion 6 are satisfied.

Restart Criterion 7

WSRC has committed to the performance of a vertical slice assessment of modifications performed on K-Reactor during the current long outage. Twenty-three modifications were chosen and, of these, six were selected for initial review. These modifications range from the 107 Diesel Generator and Distribution System to the UT Trailer addition. WSRC has also written procedure RP 5.1504, "Vertical Slice Reviews" (Reference 14) for use in this review. In this procedure, WSRC defined the make-up of review teams and defined minimum qualification criteria for the members. WSRC also defined a system for the resolution and reporting of comments and findings (if any). This review is scheduled to be completed after this evaluation has been submitted. The procedure referenced above and the plan (submitted by WSRC letter dated August 1, 1990 from C.D. Markham) submitted for the review satisfy Restart Criterion 7; however, this will remain an open item until the final engineering responses to the findings are issued and accepted.

At the present time, the vertical slice program has identified glaring deficiencies in the past design review process. The preliminary responses to the initial findings are either lacking completely or have been rejected as incomplete. A further review was performed to the modification process that governs modifications that are not line item projects. The vertical slice program has been a very successful effort, in that the deficiencies in the design review process that were suspected have been identified and documented.

Restart Criterion 8

The plan for the long-term resolution of design change control is currently being carried as ROMP item SE-5.4. The scheduled completion for the submittal of the long-term plan is mid-March, 1991. Until such time that this item is completed and reviewed, it will remain an open item.

5.4.5 Conclusion

A number of programs have been established to improve design control for the Savannah River reactors. However, during the vertical slice review program, a number of deficiencies were identified in the implementation of the design control program. DOE will continue to review the programs established to control design changes for the Savannah River reactors and report on the results of that review in a supplement to the SER.

5.4.6 References

1. WSRC Savannah River K-Reactor Technical Specifications, Rev. 0, December 18, 1990.
2. WSRC, "Quality Assurance Manual," Rev. 0.
3. Contract between Department of Energy and Westinghouse Savannah River Company, DE-AC-09-88-SR18035.

4. WSRC, RDP 12.03, "Unreviewed Safety Question Initial Screening and Safety Evaluation Procedure," Section 12.03, Rev. 0, October 26, 1990.
5. WSRC, "Design Authority Review," Reactor Technology Department Standard Practice Manual Cat 3, Sec. 8.1, Rev. A.
6. WSRC, DMP-3.01, "Reactor Restart Division Work Control Procedure," Rev. 1.
7. WSRC, "Reactor Plant Modification and Design Control Program," RDP 5.01, Rev. 0.
8. WSRC, "Technical Review of Work Packages," RP 5.1601, Rev. 0.
9. WSRC, DPSOL 105-1933, "Inspection of Seismically Qualified Systems Following Maintenance, Tests or Construction", Rev. 1, November 1, 1990.
10. WSRC, DPSOL 105-1934, "Seismic Inspection Notification," Rev. 1, October 18, 1990.
11. WSRC, "Section Level Instruction System Cognizant Engineers Desk Top Instruction Category 4," Rev. 0.
12. Institute for Nuclear Power Operations, INPO Guideline 85-031, "Guidelines for the Conduct of Technical Support Activities at Nuclear Power Facilities."
13. _____, INPO Good Practices TS-413, 85-033, "Use of System Engineers."
14. WSRC, RP 5.1504, "Vertical Slice Review, Rev. 0."

5.5 Drawing and Document Control

5.5.1 Background

The lack of adequate drawing and document control at the Savannah River Site (SRS) has resulted in incomplete or inadequate technical documentation for the production reactors, which in turn has resulted in difficulty understanding and defining their design basis. The old configuration management philosophy at SRS placed less emphasis on formal document control programs and more emphasis on select individuals with extensive SRS plant or systems experience and knowledge. SRS is currently establishing a configuration management program which will bring drawing and document control more in line with commercial nuclear industry practices.

5.5.2 Restart Criteria

The criteria for determining the adequacy of the SRS drawing and document control program for restart are described below.

- (1) WSRC shall establish and implement a formal program for control of documents and drawings. The system shall meet the criteria of the 1988 Savannah River Plant K-Reactor Restart Strategy (also known as the Charlotte Criteria) that address configuration management, design control, quality control, and records management. Formal documents shall be developed which describe the scope of the drawing and document control program and identify the types of controlled documents.

As a minimum, the controlled documents and drawings required for restart shall include (1) those design documents (e.g., calculations, drawings, specifications, and analyses) that have been developed or compiled as part of the Technical Baseline Program (TBP), (2) technical specifications, and (3) the 13 piping and instrumentation diagrams that have been validated for the central control room (CCR).

- (2) Procedures for the review, approval, and issuance of controlled documents and drawings necessary to accomplish activities and changes thereto shall be established and described to ensure technical adequacy and inclusion of appropriate quality requirements prior to implementation. Procedures shall also be established to ensure that changes to documents and drawings are reviewed and approved by the same organization that performed the initial review and approval or by other qualified, responsible organizations as delegated by WSRC.
- (3) Procedures shall be established to ensure that controlled documents and drawings necessary to accomplish activities are available at the locations where the activity will be performed prior to commencing the work and that obsolete or superseded documents and drawings are removed and replaced by applicable revisions in work areas in a timely manner. A master list or equivalent document control system shall be established to identify the current revisions of controlled documents and drawings. A formal process shall be established to update the list and distribute the updated list to predetermined responsible personnel.

- (4) Procedures shall be established and implemented to provide for the preparation of controlled as-built drawings and related documentation in a timely manner to accurately reflect the plant design.

5.5.3 Evaluation

WSRC has established a formal program for drawing and document control as discussed below.

The documentation described below establishes a program that addresses the Charlotte Criteria. Therefore, this criterion is met.

The CACMP (Reference 1) contains a broadly defined scope that includes control of documentation associated with reactor systems important to safety, documentation for other reactor systems, structures and equipment important to long-term satisfactory performance, and documentation for the reactor control room simulator. The scope also contains a table that lists the 25 K-Reactor systems important to safety and denotes the 6 initial systems to be evaluated as part of the Technical Baseline Program. The criterion to describe the scope of the program and identify the types of drawings and documents to be controlled is met.

The RRD CMPP (Reference 2) describes the specific tasks for the Drawing and Document Control Program that accomplish the scope described in the CACMP. Among the tasks is the commitment to develop and control the 13 piping and instrumentation diagrams (P&IDs) validated for the CCR prior to restart. Although the CMPP does not list the 13 safety-related systems (out of the overall 25) for which P&IDs will be developed, other reference documents list those systems. The CDL (Reference 3) indicates that in addition to Technical Baseline Documents and control room drawings, the Technical Specifications are controlled prior to restart. Therefore, the criterion to control the 13 P&IDs in the CCR, the 6 TBP design documents, and the Technical Specifications prior to restart is met.

The CACMP describes several initiatives for improving configuration management, including the development of Level I, Level II, and Level III procedures for drawing and document control. Level I procedures are those issued by the WSRC President's office. Level I procedures typically address management policies, organization, and interdivisional procedures. The drawing and document control Level I procedure is titled "Management Requirements and Procedures - Document Control" (Reference 4). Level II procedures are issued by each division manager and contain procedures used within a given division. The drawing and document control Level II procedure for the reactors is titled "RRD Administrative Manual - Controlling Documents" (Reference 5). Level III procedures are issued by a division or department manager and contain basic work instructions, guides, and standards. The drawing and document control Level III procedure is titled "Records Management Procedures Manual" (Reference 6).

The Level I Procedure reviewed contains general requirements that individual organizations (divisions and departments) that control documents and drawings (henceforth referred to as documents unless specific to drawings) which they

generate and process will develop procedures (Level II & III) for the preparation, review, approval, authentication, issuance, revision, and cancellation of those documents. When these individual groups have reviewed and approved a controlled document or a change to a controlled document, the document or revision is turned over to the Records Management Organization for distribution. The Records Management group assists individual organizations in the preparation of the lower-level procedures to ensure that procedures are uniform, that there is a good interface with the Records Management Organization and its procedures, and that applicable Quality Assurance (QA) requirements are met (Reference 7). Each organization will have a designated Procedures Coordinator who is responsible for ensuring that divisional procedures developed follow QA requirements.

The Level II Procedure reviewed provides a detailed description of how RRD will control and maintain all controlled documents for which they are responsible. In addition, processing of changes to their controlled documents is also described. Technical adequacy and quality assurance in the preparation of a document are the responsibility of each originating organization and ultimately of the authenticator of each document. Those documents designated as QA records are identified as such.

The Level III Procedures describe how the Records Management organization works with RRD and other organizations in the control, distribution, and updating of controlled documents. These procedures show a detailed process by which controlled documents are issued to Document Centers and Satellite Stations for control and distribution. The Reactor Satellite Document Control Station Desktop Guides (Reference 8) show how the staff at the Satellite Stations process controlled documents.

The criterion for developing procedures for the review, approval, and issuance of controlled documents is met. The criterion for developing procedures for the review, approval, and issuance of changes to controlled documents is also met.

The Level II procedure describes the steps used for processing controlled documents for work requests. A controlled document used as part of a work request will be identified "Working Copy" with an expiration date of 30 days (subject to re-authentication for longer projects). The requester of the working copy is placed on distribution for 30 days in order to allow notification and forwarding of document changes to the work area. The SDCS Desktop Guides show that timeliness is ensured by initiating the distribution process within four hours after receiving an approved document or revision. The criterion to develop procedures to ensure controlled documents are available at work areas is met.

A Controlled Document List has been developed. The list contains the types of documents and the divisions responsible for generating the documents and changes. More specific information such as the individual document title, revision number, and date of approval, is contained on a computer database operated by Records Management. The Level II procedure contains instructions for the establishment and maintenance of the Controlled Document List, including changes. Individual organizations will also prepare a controlled

document list of their documents, along with a controlled distribution list. These lists are, themselves, controlled and maintained by Records Management. The criterion to establish a CDL and develop procedures for updating and distributing the CDL have been met.

The RRD CMPP contains the activities used to generate the as-built drawings. The SDCS Desktop Guides contain specific procedures for the forwarding of as-built drawings and revisions, including priority forwarding to sensitive areas such as the CCR. Timeliness is ensured by initiating the distribution process within 4 hours after receiving an approved document or revision. Therefore, the criterion for timely preparation of controlled as-built drawings is met.

The RRD CMPP also contains a task for the self-evaluation of the document control program, including user compliance. While not an explicit restart criterion, this performance feedback loop gives added assurance that the drawing and document control program can be made to function as designed.

5.5.4 Conclusion

Based on the above evaluation, the DOE staff concludes that WSRC has established a formal program for the control of drawings and documents that satisfies the restart criteria.

5.5.5 References

1. WSRC, "Course of Action for the Configuration Management Program," WSRC-RP-OMM-89-020, September 1989.
2. WSRC, "Reactor Restart Division Configuration Management Program Plan", March 1991.
3. WSRC, "Reactor Restart Division Controlled Document List," October 1990.
4. WSRC, "Management Requirements and Procedures - Document Control," MRP 3.32.
5. WSRC, "Reactor Restart Division Administrative Manual - Controlling Documents," RDP 15.02, October 1990.
6. WSRC, "Records Management Procedures Manual," B-14.1, June 1990.
7. WSRC, Procedure Manual 1Q, Westinghouse Savannah River Company Quality Assurance Manual, April 1990.
8. WSRC, "Reactor Satellite Document Control Station Desktop Guides," October 1990.

5.6 Safety-Related Equipment List

5.6.1 Background

Over the years, a number of efforts to establish lists of safety-related systems and components were initiated by the operating contractors. However, a single comprehensive list using a formal classification program has never been established for SRS.

5.6.2 Restart Criteria

The criteria for determining the adequacy of the SRS safety-related equipment list program for restart are described below.

- (1) WSRC shall submit for DOE staff approval prior to restart, a program for identifying and classifying safety-related equipment in a comprehensive listing referred to as a Safety Equipment List (SEL). WSRC's program shall include a complete description of the methodology (such as safety classification definitions) for identifying and classifying the structures, systems, and equipment. Schedules, milestones, and completion dates for reactor specific SELs shall also be provided. The SEL should identify the components by system, description, tag number, and classification designation.
- (2) WSRC shall submit for DOE staff approval prior to restart, a program that defines the use of the SEL. Specifically, WSRC shall describe how the SEL is to be used for maintenance, inspection, testing activities, procurement specifications or other activities.
- (3) An interim SEL shall be submitted for DOE staff approval prior to restart. The interim SEL shall identify systems, structures, components, subsystems, and system boundaries, but need not include the safety classification designations. The interim SEL shall include the following systems:
 - o Top Priority Systems: WSRC shall produce an interim SEL for the 13 safety systems, which were selected by the Reactor Technical Baseline Program (see Table 5.6-1).
 - o Other Systems: WSRC shall produce an interim SEL for the 12 remaining systems and structures identified in Table 5.6-1. Identification of the system components and system safety related boundaries for the interim SEL may be limited to currently identified components and boundaries.

WSRC shall meet the following criterion post-restart:

The program for identifying and classifying equipment important to safety shall be implemented on the schedule approved.

5.6.3 Evaluation

Attachment 1 of OPS-DOE-90-016 (Reference 1), "Criteria for Classification of Structures, Systems and Equipment Important to Safety for SRS Production Reactors," described the methodology to be used for identifying and classifying systems and equipment important to safety. The methodology described consisted of seven techniques. One of these techniques was established to allocate safety classification designations, as required by the restart criteria. The other six techniques were established to categorize equipment per different schemes, such as American Society of Mechanical Engineers (ASME) codes, Institute of Electrical and Electronics Engineers (IEEE) codes, quality assurance, seismic and environmental qualification, and procurement levels. They are, therefore, outside the scope of this evaluation.

The safety classification methodology was derived from the standard, ANSI/ANS-51.1, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants" (Reference 2). The plan describes three safety-related classes and one nonnuclear safety class. The definitions for Safety Classes 1, 2, and 3 were found to comply with the standard definitions contained in 10 CFR 50.55a and NRC Regulatory Guide 1.26 (References 3 and 4). The definition for the nonnuclear safety class was included, but is outside the scope of this evaluation.

Attachment 2 of OPS-DOE-90-016, "Schedules, Milestones, and Completion Dates for Generating Reactor-Specific Safety-Related Equipment List," provides the schedules for the reactor-specific SELs, associated milestones, and completion dates.

The SEL schedule is consistent with the schedules of other programs designed to use the SEL. In particular, the Vendor Manual Program for K-Reactor is presently scheduled for completion by December 1993. The final SEL for K-Reactor is expected to be completed by June 1993. The schedule for the Work Management System for K-Reactor is also consistent with the SEL schedule for K-Reactor.

Attachment 3 of OPS-DOE-90-016, "Safety-Related Equipment List Utilization," provides the controls over maintenance, procurement, inspection and other activities. The descriptions include the computerized information database; the process by which the procedures applicable to maintenance, parts replacement, and other activities would be determined; and the management controls used to verify that the appropriate procedures will be followed. A description of the methods for design verification of safety-related parts procurement and for maintaining a complete and current vendor information program is also provided. These practices were developed using guidance from commercial standards (References 5 and 6) and were found to be adequate in complying with the restart criteria.

OPS-DOE-900039 (Reference 7), "K-Reactor Interim Safety Equipment List," provides the interim SEL for K-Reactor. This document consists of a detailed listing of all safety-related components for the 13 top-priority systems. Existing central control room drawings and critical component lists were used to verify the completeness of this document. Approximately 200 components

were identified from these two sources, using a statistical methodology. All of them were determined to be addressed adequately by the interim SEL. The remaining 12 systems, including the confinement system, have been analyzed on a more limited basis in accordance with the restart criteria.

The format of the interim SEL is consistent with the requirements of the restart criteria. This format can be expanded to include the additional information required for the final SEL.

5.6.4 Conclusion

Based on the above evaluation, the DOE staff concludes that WSRC has developed an acceptable safety-related equipment list program that satisfies the restart criteria.

5.6.5 References

1. Safety Equipment List, OPS-DOE-90-016, June 1, 1990.
2. American National Standards/American Nuclear Society, ANSI/ANS 51.1-1983, April 1983.
3. Title 10, Code of Federal Regulations, Part 50.55a, 1989.
4. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.26, February 1976.
5. Title 10, Code of Federal Regulations, Part 50, Appendix B, 1989.
6. American National Standards/American Nuclear Society, ANSI/ASME-NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities," 1986.
7. K-Reactor Interim Safety Equipment List, OPS-DOE-900039, November 28, 1990.

TABLE 5.6-1

SYSTEMS TO BE EVALUATED FOR THE SAFETY RELATED EQUIPMENT LIST

Baseline Systems

Reactor Scram System (including safety rods)
Process Water Diesel Generators
Cooling Water System
Emergency Power System
Process Water System
Control Rod System
Central Control Room Habitability
Supplementary Safety System
Water Removal and Storage
Emergency Cooling System
125 V DC Power
Discharge Assembly Cooling
Reactor Room Spray System

Other Systems

Airborne Activity Confinement System
Monitoring and Actuation System
Moderator Recovery System
120 VAC Power
Confinement Heat Removal System
Process Water Pump and Motor Lubrication Oil System
Structures/Buildings (as defined by the SEL to be safety-related)
Compressed Air
Disassembly Basin Cooling and Filtration System
Blanket Gas System
Reactor Tank Pressure Relief System
Reactor Tank Internals and Supports

5.7 Risk and Reliability Analysis

5.7.1 Background

The principal tool supporting this activity is the full-scope Probabilistic Risk Assessment (PRA) that WSRC is developing for the SRS reactors. The PRA, when completed, will include Level 1, 2, and 3 analyses. The Level 1 analysis identifies core damage sequences and their frequencies. The Level 2 analysis studies each accident sequence from core melt progression, through confinement system response, to the release of radionuclides to the environment. The Level 3 analysis takes the release from each accident sequence and estimates the consequences. Accident initiators include internal events, which are associated with component failures in the reactor systems, and external events, which are outside the reactor systems, such as earthquakes, fire, or flooding.

The Level 1 analysis has been completed, and the final Level 1, Rev. 0, report has been published (Reference 1). Completion of the Level 1, Rev. 0, PRA is a requirement for reactor restart. It should be noted that the Level 1, Rev. 0, PRA is based on the 1987 plant configuration and that a Level 1, Rev. 1, PRA based on the restart plant configuration is expected in the Fall of 1991. The Level 2/3 analysis has only recently been completed, with publication of the Level 2/3, Rev. 0, draft report expected shortly thereafter. The Level 2 and 3 analyses are not required to be completed before restart, but will be evaluated as part of the ongoing PRA review process.

The SRS PRA undergoes extensive reviews as part of its Quality Assurance process. WSRC reviews are conducted by members of its own technical staff and by the Senior Review Panel, whose members are recognized experts in PRA. These WSRC-sponsored review groups provide feedback to the PRA staff on all aspects of the PRA.

DOE reviews the SRS PRA as part of the independent review process. An ongoing, detailed review of the SRS PRA is conducted by the DOE Independent Review Panel (IRP), a group of PRA practitioners. The IRP has conducted a preliminary review of the Level 1, Rev. 0, report (Reference 2) and will complete their review of the Level 1, Rev. 0, PRA prior to restart. A one time, high-level review of the Level 1 PRA was conducted by the DOE High Level Peer Review Committee, which consisted of nationally recognized PRA experts as well as some members of the IRP. In their report (Reference 3), the High Level Peer Review group stated that "The Committee does not see any safety issues resulting from the SRS PRA that would be cause for delaying restart of the K-Reactor at the Savannah River Site."

External independent oversight has been provided by the Advisory Committee for Nuclear Facility Safety (ACNFS) and the Defense Nuclear Facility Safety Board (DNFSB). This SER is based on a review of the Level 1, Rev. 0, report, the results of reviews performed to date, and several related communications as noted.

5.7.2 Restart Criteria

Methodology

The methods selected for the SRS reactor PRA shall be supported by technically sound and defensible bases. The analyses shall be predicated on state-of-the-art techniques in the areas of:

- o initiating events and success criteria
- o event tree modeling
- o fault tree modeling
- o data analysis
- o human reliability analysis
- o analysis of dependence such as common cause events
- o accident sequence quantification
- o uncertainty, sensitivity, and importance analyses

All potential internal and external events shall be considered and analyzed in the PRA. The PRA shall be in sufficient detail and depth to identify plant-specific vulnerabilities that may affect reactor restart and operation.

Documentation

Documented information shall be presented clearly and concisely in a hierarchical fashion, ranging from general to specific information. The documentation of plant and system performance, accident sequence progression, and PRA results shall be presented in a form that can be understood by DOE and contractor decision makers.

Specific Criteria

The criteria for determining the adequacy of the SRS risk and reliability analysis in safety design for restart are the following:

- o A Level-1 PRA, meeting the applicable criteria listed above, shall be completed prior to restart.
- o Sufficient probabilistic information shall be developed to support the Environmental Impact Statement (EIS).
- o The PRA application program shall be effective and well-structured for supporting the reactor restart effort.
- o A mechanism for feedback on PRA recommendations shall be developed for DOE and contractor decision makers.
- o The PRA Program shall prioritize Reactor Safety Improvement Program (RSIP) work tasks on the basis of risk-based insights and shall be used selectively for restart activities.
- o A PRA Applications Group shall be created to address long term issues and to work intimately with the PRA staff.

- o A formal process shall be developed and implemented for the involvement of the PRA Applications Group in the resolution of long term issues.

5.7.3 Evaluation

This section presents DOE's evaluation of the Level 1, Rev. 0, PRA. Section 5.7.3.1 discusses the methodology used in the PRA. Section 5.7.3.2 discusses the documentation of the results in the PRA report, and Section 5.7.3.3 discusses the PRA applications activities. The compliance with each of the specific criteria for restart is summarized in Section 5.7.3.4.

5.7.3.1 Methodology

Internal Events

Initiating Events and Success Criteria

The PRA methodology for the Level 1 analysis of internal initiating events generally follows the "linked fault tree" approach as defined in NUREG/CR-2300, "PRA Procedures Guide," (Reference 4). For the development of a complete set of initiating events, a hierarchical decomposition by means of logic trees called "dendograms" was performed. This unique approach was complemented by examining SRS reactor-incident data and by a review of the EPRI initiating event list for commercial nuclear reactors. Although there is no one method that guarantees the completeness of any initiating event list, the PRA Level 1 report shows that a systematic search was performed. Therefore, the methodology for determining initiating events is acceptable.

Success criteria were extracted from information contained in plant procedures, supplemented by the PRA team's experience with the plant; reviews of existing SRS engineering/physics analysis; and analyses done in support of the PRA. Additional thermal-hydraulic analysis to refine some of the success criteria is planned for 1991. It is expected that the impact of improved success criteria will be a reduction in the estimated core damage frequency. Therefore, this issue does not require resolution before restart.

Event Tree Modeling

The reviews of the event trees indicate that the trees accurately model the responses to the postulated initiating events, given the simplifying assumptions used to reduce the complexities of the analysis. The principal scoping assumption, i.e., that the reactor is operating at full power, is typical of most reactor PRAs. Some concern was expressed that shutdown operations and operation at less than full power were not modelled. Commercial reactor experience has shown that analyses of these operational phases can provide important insights. These concerns need not be resolved before restart, because it is expected they will have only small contributions to core damage frequency and plant risk.

Fault Tree Modeling

The system models consist of system-level fault trees and detailed generic-component fault trees. The component fault trees are used to model component-specific failures and system dependencies. The PRA team constructed the fault trees in such a manner that they can be used for the same function but with different success criteria. Thus, one fault tree can be used for the same function in different accident sequences. This was accomplished by the extensive use of flags. This method of system analysis is acceptable.

Data Analysis

Plant-specific data were used as much as possible for the development of failure rates. Generic data were used in limited cases when believed to be the better data source. No combining of the two data bases was performed.

The review teams were generally impressed with the data effort; however, the PRA report does not provide sufficient evidence that a thorough data-trending analysis (in particular, the potential effects of aging) was performed. Therefore, there was some concern that failure probabilities of some equipment could be higher than calculated. This is a long-term concern that could be addressed by the establishment of an equipment surveillance program that would assure the early identification and correction of equipment degradation caused by aging. This program is part of the PRA applications plan and is not a restart item.

Another concern was the statistical treatment of equipment that has not failed. In these cases, an effective value of 0.33 (1/3 of a failure) is assumed for purposes of estimating the failure rate. For highly reliable components, this approach may result in considerable overestimation of the actual failure rates. For instance, the initiating event frequency for failure of the bellows (the expansion joints that connect the primary and secondary cooling lines to the heat exchangers), which results in a loss-of-coolant accident (LOCA), is calculated using this approach. Failure of a bellows is a dominant contributor to the core damage frequency in the present PRA. If the frequency of the bellows-LOCA is estimated too conservatively, it could result not only in an overestimate of the overall core damage frequency, but also might mask other contributors that should be thought of as important to core damage frequency.

A long-term initiative should be to reanalyze the failure rates of risk-important components that have no previous failures using state-of-the-art mechanistic techniques. Ideally, the techniques applied will be capable of addressing design robustness, enhanced inspection, and condition-monitoring of components. Although the risk profile may be different when more realistic data are used, the overall estimated core damage frequency and risk will most likely be smaller. Therefore, this issue does not require resolution before restart.

Human Reliability Analysis

Human actions are important contributors to safety; therefore, it is important to perform a realistic human response analysis (HRA) using state-of-the-art methodology (Reference 5). The predominant technique used in the PRA was the THERP method, which is appropriate when the operators do not have extensive diagnostic tasks to perform.

However, it is known that SRS is developing symptom-based procedures to augment the existing event-based procedures. Symptom-based procedures imply that there are conditions under which diagnosis by the operators is required. Under such circumstances, THERP analysis may not lead to accurate answers, especially when independence of multiple human actions is assumed. This could lead to underestimation of sequence frequencies. The PRA team plans to revise the analysis using state-of-the-art techniques, including simulator testing, and to address the issue of dependencies among operator actions. The revised analysis is not necessary for restart decision-making, but will be needed when value-impact analyses are being performed as part of the justification of future system improvements.

Analysis of Dependence Such as Common Cause Events

Intersystem dependencies (support-system dependencies on AC electric power and cooling water) are modeled in a satisfactory manner. Although the modeling of other support systems, such as DC power, is not as comprehensive, and some support systems, such as instrument air, are not modeled, the PRA team made judgements that comprehensive analyses were not necessary.

The common cause analysis for system failures is performed in a generally systematic manner. To quantify the effects of common cause failures, a beta factor approach was used. Conservative screening values of 0.1 for two failures (0.05 for three or more failures) were used for the beta factors. More-realistic estimates for the beta factors will be developed for long-term risk management applications to ensure acceptable value-impact analyses. Despite the coarseness of the present estimates, it is not believed that more accurate values would result in significant new engineering insights. Therefore, this issue does not require resolution prior to restart.

In support of SRRO, an independent systems analysis was conducted by the Los Alamos National Laboratory (LANL), which identified potential common cause failures and answered specific questions regarding system-dependency vulnerabilities. The results of the LANL efforts generally supported the results of the WSRC PRA.

LANL provided 27 engineering insights to SRRO in support of restart (Reference 6). All 27 insights have been or are being addressed by restart or post-restart programs. One insight identified a potential contributor to the core damage frequency. The insight questioned whether the process water pump diesel generators (CAT diesels), which supply DC power to the process water pump DC motors, would be adequately cooled during a station blackout condition (loss of all AC). WSRC had previously addressed this insight as part of its rebaselining program by conducting an engineering evaluation of cooling water

flow to the diesels. Based on the LANL evaluation, the high-level review panel recommended testing of the CAT diesel cooling system prior to reactor restart (Reference 7). This testing will be accomplished as part of startup testing and will provide added assurance that the process water pumps will operate during a station blackout. LANL will continue its support role of providing independent PRA analyses.

Accident Sequence Quantification

The quantification process follows the large fault-tree linking method (Reference 4). The unavailability and unreliability values for the electric power system are obtained from a Markov model, which requires the setting of flags to turn parts of the linked model off and on. The review teams expressed concern about the complexity of the quantification process and the fact that several computer programs used by the PRA staff for the quantification have not been validated. However, no problems have been found, and the independent systems analysis by LANL has shown reasonable agreement with the SRS PRA.

Uncertainty, Sensitivity, and Importance Analyses

An important feature of PRAs is the presentation and display of various statistical results that are produced from uncertainty, sensitivity, and importance analyses. Although cutsets and importances could not be provided for the full model, review by DOE indicated that the analyses are acceptable.

External Events

The Rev. 0 report includes a discussion of the seismic analysis methodology and results. DOE has determined the seismic risk methodology for integrating seismic hazard and fragility data and the methodology for quantifying the seismic risk are conventional and satisfactory.

The fire analysis estimated the total core damage frequency from fires to be only 1.4×10^{-7} per year. This estimated core damage frequency is low when compared to results from fire analyses for commercial nuclear power plants. While the method used for the internal fire analysis is conventional, the SRS reactor design (low temperature, wide separation of areas, DC flow independent of the CCR) is not, and may contribute to the low core damage frequency. Although some review groups had concerns (Reference 8), it is not anticipated that resolution of any of these concerns would result in fire becoming an important contributor to the total estimated core damage frequency. Therefore, these concerns do not require resolution before reactor restart. These concerns (and all others) will be followed as part of the continuing review process. Appendix R and conventional fire prevention SER restart criteria require separate review. The PRA models and results will be used in support of these separate reviews and any decisions for restart fire protection requirements as appropriate.

5.7.3.2 Documentation

Internal Events

The documentation of the internal events analyses is adequate for restart. Improvement could be made in the following areas:

- o Traceability of the P&IDs and other system information
- o Dependency matrices or tables
- o Cross-referencing between the system descriptions and the PRA data
- o The bases for the success criteria
- o Cutsets and importance values for the main analysis instead of only for the simplified model

External Events

The documentation of the external events analyses is adequate. It was not clear which of the four seismic analysis cases should be regarded as the "base case," although this is not a consequential problem. The next revision of the Level 1 PRA will be based on K-Reactor in the restart configuration.

5.7.3.3 Applications

WSRC has published the Safety Information Document (SID) (Reference 9) that provides backup safety information to support the Environmental Impact Statement (Reference 10). Included in the SID is a chapter that presents estimates of reactor risk based on the restart configuration. The PRA information in the SID was reviewed by members of the IRP, and meetings were held between the IRP and PRA staff to discuss the results and the analyses performed. Initial concerns brought up by the IRP (Reference 11) were addressed, and the results were determined to be acceptable for use in the EIS.

A formal Risk Management (RM) process is under development at WSRC. WSRC's commitment to PRA and active RM is strong. The staff size related to PRA, RM, and other activities is substantial and growing. The range of experience and general quality of the staff ensures competent results.

Key PRA managers sit in on Restart Issue Management Program (RIMP) committee meetings. These managers review each restart issue and provide a risk perspective. Although the PRA group is not formally tied into the decision process until an issue reaches the RIMP, the PRA staff has several means of including issues in the decision process. The PRA Manager and the Risk Application Group Manager also review RSIP tasks for prioritization.

The PRA staff has issued "green letters" (WSRC's vehicle for introducing issues and recommendations for formal evaluation) to the Reactor Restart Division. Reference 12 provided recommendations for improved procedures and

training related to operator actions that mitigate severe accidents. Reference 13 provided recommendations regarding the frequency of surveillance testing and allowable out-of-service times for certain reactor systems. These letters and others have been evaluated by the Reactor Engineering Department, which has responded in writing. This process has resulted in the majority of the green letter recommendations being addressed. The PRA group also interacts with the restart program through personal relationships and the promotion of PRA by several managers. As a result, most issues that result in an implementation project are evaluated by the PRA staff (Reference 14).

The mechanism to include PRA results and insights in the restart program is satisfactory. The PRA group is involved in decisions that lead to the elimination of issues from the restart program, although issues may be added to the restart program without involvement of the PRA staff. The PRA group has provided input to the restart effort on seismic and fire issues, and guidance for operator training on the simulator. The staff has also developed a prioritized list (ranked by importance to core damage frequency) of components for the restart test team. A PRA Applications Group was created in June 1989 to address long-term issues. The group works intimately with the PRA staff and addresses restart issues as requested. The PRA Applications Group has developed an ambitious program. Of primary importance is the establishment of formal mechanisms that assure PRA oversight and evaluations of operational events and design modification activities. For instance, PRA perspectives are used to evaluate public safety attributes used to prioritize activities evaluated under the Reactor Safety Improvement Program. An Applications Program Plan (Reference 15) was completed in March 1991.

5.7.3.4 Compliance with Specific Restart Criteria

- o "A Level-1 PRA, meeting the applicable criteria listed above, shall be completed prior to restart."

WSRC has complied with this criterion.

- o "Sufficient probabilistic information shall be developed to support the EIS."

WSRC prepared the SID, based on the PRA as it existed in 1989, to provide requisite information for the EIS. This was reviewed by the IRP. WSRC addressed all the IRP findings and has complied with this criterion.

- o "The PRA application program shall be effective and well-structured for supporting the reactor restart effort."

WSRC has complied with this criterion.

- o "A mechanism for feedback on PRA recommendations shall be developed for DOE and contractor decision-makers."

WSRC has complied with criterion.

- o "The PRA program shall prioritize RSIP work tasks on the basis of risk-based insights and shall be used selectively for restart activities."

The WSRC PRA staff is involved with the RIMP process and the RSIP process at this time. Therefore, WSRC has complied with this criterion.

- o A PRA Applications Group shall be created to address long-term issues and to work intimately with the PRA staff."

WSRC has complied with this criterion.

- o A formal process shall be developed and implemented for the involvement of the PRA Applications Group in the resolution of long-term issues."

WSRC has complied with this criterion.

5.7.4 Conclusions

Methodology

On the basis of this review, the staff concludes that, in general, the PRA employs state-of-the-art methodology. The methodology is acceptable for restart purposes. Even though the methodology described in the Rev. 0 report is acceptable for restart purposes, there are several areas where improvements could be made. The improvements that WSRC is planning to accomplish in 1991 should provide an even more accurate PRA model of the reactors.

Documentation

The overall organization of the Rev. 0 report is greatly improved over the draft report that was issued in 1989. It is considered to be acceptable for restart purposes.

Even though the Rev. 0 report is deemed acceptable for restart purposes, it is still difficult to review and understand. The work planned by WSRC for 1991, which will culminate in a Level 1, Rev. 1, report, should be more understandable.

Applications

The role of the PRA in the restart effort is deemed satisfactory. The involvement of the PRA Applications Group in the resolution of long-term issues is being formalized. This will ensure that such evaluations are correct and consistent with the current PRA model. The Applications Program Plan indicates that WSRC has formalized its risk application approach. Currently, the risk application is handled by both the PRA Methodology Group and the Risk Application Group. As the Risk Application Group increases its staffing levels, it will be able to more fully integrate itself into Risk Management activities. It is expected that the PRA Methodology Group will always maintain some level of involvement in risk applications.

5.7.5 References

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2. Letter from H. Reilly to K. Murphy, "Preliminary Review of Rev. 0 PRA Report," February 1991.
3. B. J. Garrick et al.(DOE-DP), "Savannah River Site Probabilistic Risk Assessment High-Level Review," DOE-DP, February 1990.
4. NUREG/CR-2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," Vol. 1 and 2, January 1983.
5. U. S. Nuclear Regulatory Commission, NUREG/CR-1278, "Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications," August 1983.
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7. Memo from B. J. Garrick to Kenneth G. Murphy, "LANL Meeting And A Discussion Of Their Engineering Insights Report On The SRS PRA," April 11, 1990.
8. EG&G Idaho, Inc.; Pickard, Lowe and Garrick, Inc.; and Sciencetech, Inc., "Detailed Review of the Draft Savannah River Reactors Level 1 PRA," March 21, 1990.
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10. DOE/EIS-0147, "Final Environmental Impact Statement: Continued Operation of K-,L-, and P-Reactors, Savannah River Site, Aiken, South Carolina," December 1990.
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14. H. E. Filacchione, SCIENTECH Trip Report to Tony Eng, Re: Report on Meeting with WSRC PRA Staff, October 31, 1989. H. E. Wingo and the High-Level Review Committee, personal communication, January 12, 1990.
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5.8 Software Quality Assurance (SQA)

5.8.1 Background

Historically, the Savannah River Laboratory (SRL) has not had a formal software quality assurance program comparable to those found in the commercial nuclear industry. In 1983, the Nuclear Regulatory Commission (NRC) required that licensees of commercial nuclear reactors electing to perform their own safety analyses in support of licensing actions must demonstrate their ability to perform the analysis through code verification since that capability involves the setup and execution of complex computer codes.

For a similar reason, DOE committed to a requirement that safety-related computer codes, which are used in safety analyses affecting the Safety Analysis Report (SAR), Technical Specifications, Technical Standards, Mechanical Standards, Technical Manuals, and Reactor Operation Procedures, be identified, verified, validated, and certified. The personnel who perform safety analyses are also required to be certified. Computer models included in safety analyses are required to conform to Quality Control and Quality Assurance procedures and to be controlled and referred to in new or revised safety analyses.

5.8.2 Restart Criteria

The criteria for determining the adequacy of the SRS Software Quality Assurance for restart are described below.

(1) Applicability

Safety-related software is defined as those computer codes used in safety-related analyses done in support of the Technical Specifications, SAR, and Probabilistic Risk Assessment (PRA). This limited set of software is selected to provide reasonable quality assurance for the technical programs essential to the reactor operation safety.

(2) Software Life Cycle

WSRC shall submit to DOE a plan to meet the criteria listed below.

The software documentation (Item 5 below) required to enable a qualified user to perform the applicable analyses independent of any additional information, along with a record of all code modifications, shall be readily available for the life of the reactors. Any significant differences between the results of standard test problems used to analyze software and the analysis results on record shall be documented with explanations where available and submitted to DOE for review.

The Software Life Cycle may include a retirement phase, which shall contain provisions to ensure that the retired software is not changed, that configurations are adequately documented, and that retired software is not made available for unrestricted use.

(3) Software Verification and Validation

Software verification and validation activities and schedule shall be as described in the "Task Assessment Schedule for Certification of Reactor Analysis Computer Codes (U)" WSRC-TR-90-397 (Reference 1). A plan for activities and schedules for SAR and PRA codes similar to WSRC-TR-90-397 shall be provided by the operating contractor. This plan shall be reviewed and approved by DOE prior to restart.

There are three types of testing problems that may be used to validate computer codes:

- (a) Problems to which the answer is known exactly. (These problems tend to help correct the larger code errors since the correct answer is known.)
- (b) Problems that can be, and have been, analyzed using other codes. (These problems help identify errors involving empirical correlations.)
- (c) Problems for which there is an experimental measurement of the right answer. (These problems tend to raise the confidence of the user by also checking empirical correlations as well as the code's operation.)

A combination of (a) and (c) or (b) and (c) are acceptable for the purpose of code validation. For applications described above where test problems (c) are not available, the use of (a) and/or (b) is acceptable.

Verification and validation reviews requirements shall be developed and implemented by WSRC in compliance with ANSI/ASME NQA-2 (Reference 2). Deviations from these requirements shall be documented, justified with compensatory actions, and submitted to DOE for review.

Verification reviews shall identify the participants and their specific responsibilities during the review, preparation, and distribution of the review documentation.

The review documents shall be updated and placed under configuration control.

Documentation of review comments and their disposition shall be retained until they are incorporated into the updated software. Comments and their disposition not incorporated shall be retained in accordance with established procedures.

The review of software requirements shall be performed at the completion of the software requirements documentation. This review shall ensure that the requirements are complete, verifiable, and consistent. The review shall also ensure that the requirements will result in feasible and usable codes.

The software design review shall be held at the completion of the software design documentation. This review shall meet the design verification requirements of NQA-1, 3S-1. This review shall evaluate the technical adequacy of the design approach, internal completeness, consistency, clarity, and correctness of the software design, and shall verify that the software design is traceable to the requirements.

Upon completion of the testing phase (and the installation phase if necessary) the development cycle documentation shall be reviewed and approved to ensure completion and acceptability.

(4) Software Configuration Control

WSRC shall submit to DOE a plan to meet the criteria listed below.

WSRC shall develop and implement configuration control requirements in compliance with ANSI/ASME NQA-2. Deviations from these requirements (i.e., for older software) shall be documented, justified with compensatory actions, and submitted to DOE for review.

(a) Configuration Identification

- o A configuration baseline shall be defined at the completion of each major phase of the software development. Approved changes created subsequent to a baseline shall be added to the baseline.
- o A labeling system for configuration items shall be implemented that: (1) uniquely identifies each configuration item, (2) identifies changes to configuration items by revision, and (3) provides the ability to uniquely identify each configuration of the revised software available for use.

(b) Configuration Change Control

- o Changes to software shall be formally documented. This documentation shall contain a description of the change, the rationale for the change, and the identification of affected baselines.
- o The change shall be formally evaluated and approved by the organization responsible for the original design, unless an alternate organization has been given the authority to approve the changes. Only authorized changes shall be made to software baselines. The revised software baseline must be verified and validated as necessary. It must be verified that the change is appropriately reflected in software documentation and that document traceability is maintained. Software changes shall also be reviewed as part of the design change control and unreviewed safety question processes.

(5) Documentation

Documentation shall be established in accordance with the schedule described in the "Task Assessment Schedule for Certification of Reactor Analysis Computer Codes (U)," WSRC-TR-90-397. Documentation for safety-related software shall be maintained for the life of the reactors. That documentation shall include verification and validation and configuration control documentation as discussed above and meet the requirements of ANSI/ASME NQA-2. Where full compliance with ANSI/ASME NQA-2 is not met, the following documentation shall be maintained:

- (a) software application
- (b) input data description
- (c) method of solution
- (d) output data description
- (e) statement of accuracy and limitations
- (f) source listing
- (g) sample run
- (h) description of access control and security
- (i) user qualification requirements
- (j) manuals and supporting documentation
- (k) documentation of reviews
- (l) record of updates and discrepancy resolutions

(6) Problem Reporting and Corrective Action

A formal software problem procedure and corrective action that complies with ANSI/ASME NQA-2 shall be established for software errors and failures. This problem reporting system shall ensure that problems are promptly reported to affected organizations to facilitate formal processing of problem resolutions.

WSRC shall establish and implement a classification system for identified problems. The classification system shall have defined criteria based on the impact of the software output.

Corrective action by the responsible organization shall ensure that:

- (a) Problems are identified, evaluated, documented, and, if required, corrected.

- (b) Problems are assessed for impact on past and present applications of the software by the responsible organization.
- (c) Corrections or changes are controlled in accordance with the configuration change control requirements discussed above.
- (d) Preventive actions and corrective actions results are provided to affected organizations.

(7) Access Control

Access control shall be established to permit authorized access and prevent unauthorized access to a computer system.

(8) Software Procurement

Software procurement practices shall meet the requirements prescribed by the ANSI/ASME NQA-2 standard.

Individuals or organizations developing and supplying software under contract shall be required to have policies and procedures that meet the applicable requirements of ANSI/ASME NQA-2 as specified in procurement documents. The documentation that is required by ANSI/ASME NQA-2 shall be delivered or made available by the supplier to WSRC. The applicable requirements of ANSI/ASME NQA-2 shall become the responsibility of WSRC upon receipt of software. Typically, this software enters WSRC's organization at the start of the installation and checkout phase. The supplier shall report software errors or failures to WSRC, and the WSRC shall report software errors to the supplier.

Software that has been developed not using this standard shall be placed under the configuration controls required by the ANSI/ASME NQA-2 standard prior to use. For this software, WSRC shall perform and document an evaluation to:

- (a) Determine its adequacy to support software operation and maintenance.
- (b) Identify the activities to be performed and the documents needed for the software to be placed under configuration control. This determination shall be documented and shall identify as a minimum:
 - o user application requirements
 - o test plans and test cases required to validate the software for acceptability
 - o user documentation as required by Documentation Section 5.8.2.5.

After the specified activities are performed, reviewed, and approved, the software shall be placed under configuration control.

As an alternate, WSRC shall obtain the above documentation from the supplier or perform a documented review of the documentation at the supplier facility to determine acceptability. This review shall meet the requirements as specified above.

For existing or new codes not developed or modified onsite, the contractor shall determine and obtain a level of confidence in the code appropriate to its use. A plan to do this in a rigorous fashion shall be submitted to DOE for approval after restart.

The organization providing software services shall have a plan(s) for software quality assurance that meets the requirements of the Software Procurement section of NQA-2. WSRC shall determine the adequacy of this plan.

(9) Records

Compliance with the above requirements shall be documented and maintained in the QA record system unless an acceptable (approved) explanation for any omission is provided.

(10) Certification

WSRC shall provide an official written statement for all certified computer codes stating that all above requirements and all SRS Quality Assurance Manual software control requirements have been met. The written statement shall also state that WSRC has established implementation procedures for the use of computer codes by certified personnel.

For each code user, WSRC shall maintain an official written statement attesting to the user's competence based on the user's familiarity with applicable quality assurance requirements.

5.8.3 Evaluation

The National Academy of Science (NAS) recommended (Reference 3) that "...management of software engineering activities should be reviewed and improved." Relating to this recommendation, the Reactor Operations Management Plan (ROMP) (Reference 4) QA 19.1 requires an independent assessment of all restart-related process and research software activities by recognized software quality assurance (SQA) experts. The primary objective of the assessment was to evaluate the technical and management controls used to design, test, install, operate, maintain, and retire software having high impact on safety, environment, security, production, material process and material accountability. The resulting Reactor Restart Software Quality Assurance Assessment Final Report (Reference 5) indicates that "the current QA procedure 20-1 requirements will satisfactorily meet appropriate industry standards for WSRC with the addition of specific requirements for configuration management, software evaluation, design documentation, and independent verification and validation." It also indicates that "an independent assessment of the implementation of management plans to meet SQA

Program requirements is recommended to ensure that actions taken are effective and adequately address the specific observations identified." The staff concurs with and endorses this recommendation.

NAS recommended that "the Department of Energy should commit to a significant program of severe accident model development and validation." The restart criteria define the scope of applicability of the criteria to three safety programs and their associated codes: Technical Specifications, SAR, and PRA. The computational tools developed in the Severe Accident Assessment Program (SAAP) are used in PRA consequence analyses; therefore, the validation of these codes is covered implicitly in the "Software Verification and Validation" section of this SER.

Applicability

A detailed plan and schedule (Reference 6) for completion of the Verification and Validation (V&V) program for computer codes by WSRC Nuclear Reactor Technology and Scientific Computations (NRTSC) was submitted to DOE for review. The V&V plan includes 12 reactor physics codes and 9 thermal hydraulics codes. Four reactor physics codes (GILDA, GRIMHX, GLASS, and TRIMHX) support the bases of the new Technical Specifications: Safety Rod Insertion Limit, and Safety Rod Operability. Three thermal hydraulics codes (TRAC/MOD1, FLOWTRAN-FI, and RELAP5) and six reactor physics codes (AA3, GILDA, GRIMHX, WIGGLE, SHIELD, and TRIMHX) support the thermal hydraulics limits program. The V&V plan did not include codes to fully support the SAR or the PRA. SAR and PRA code V&V and certification plans did not include consequence and engineering analysis codes, such as MACCS and CONTAIN, supporting the environmental impact statement (EIS) and the Safety Information Document (SID). This open item is being addressed in the exemption request and demonstration of confidence proposal document prepared by SRL/SARM for DOE review and approval. This work is bounded by ROMP work scope as reported in WSRC memo OPS-RRP-900156. The exemption request is being reviewed by SRSP0 and SRRO for approval.

On January 15, 1990, the Certification Plan for Reactor Analysis Computer Codes (Reference 7) was submitted to DOE for review. This plan addressed certification of the 21 codes included in the V&V plan of November 15, 1989.

Software Life Cycle

The Certification Plan for Reactor Analysis Computer Codes identifies milestones for completion of generalized configuration control procedures. These procedures are as follows:

- o moving a code into configuration control
- o operating under configuration control
- o changes that proprietors make
- o changes in computer environment
- o archiving responsibilities and backup disaster control
- o retired codes
- o tracking of configuration control documentation
- o error identification and correction procedures

- o generalized training requirements for users and operators
- o periodic code testing with test problems
- o use of work stations and PCs in critical calculations

The plan does not address code re-certification when updates are made to the codes. In a WSRC letter (Reference 8) WSRC has agreed that code re-certification will be addressed as the certification plan is implemented. This is an open item.

Software Verification and Validation

The objectives of the V&V plan of November 15, 1989, meet the restart criteria for software verification and validation, except that the plan lacks the necessary measures for corrective actions once deficiencies are identified during V&V. This is an open item. The plan objectives are:

- o Identification of computer codes requiring verification and validation.
- o Development of benchmark packages, including standard problems, to ensure proper definition of the requirements for portability and limitations for the codes.
- o Documentation of an experimental basis, where practical, for the code calculations and linking the results to tests conducted.
- o Ensuring that user documentation is complete and identified.

The V&V plan provides a V&V action matrix for each code, which includes verification review and validation/benchmarking review. The plan indicated that the completion of this milestone is scheduled for September 30, 1991. An exemption request was submitted for DOE approval (on May 23, 1990) which documented the short term deviation from the V&V requirements and the necessary justification with compensatory actions. The request covered the Nuclear Reactor Technology and Scientific Computations codes used in support of the Technical Specifications and the Power Limits programs. The short-term and long-term strategies outlined in the request are consistent with an earlier approach dealing with the approval of Technical Specifications important to reactor shutdown capability (Technical Specification Parameter Justification) where important shutdown parameters and the associated analysis software were identified. A demonstration of confidence in the software was required as condition for approval of these restart Technical Specifications. Furthermore, in order to fully satisfy the restart criteria for code certification, benchmarking reports for the RELAP, TRAC, and FLOWTRAN codes using test data identified in the request would be developed. It was also understood, as indicated in the request, that a similar approach would be developed for Probabilistic Risk Assessment (PRA) and Safety Analysis Report (SAR) codes. The request was approved by DOE on July 17, 1990 (Reference 9).

Since the draft V&V and Certification plans for SAR and PRA codes (References 10 and 11) show a completion date beyond restart, a similar exemption request is also necessary for PRA and SAR codes. This is an open item. In a

SRSP0/WSRC interaction meeting on December 8, 1989, the Safety Analysis and Risk Management (SARM) manager agreed to provide a demonstration of confidence in the SAR and PRA codes as a restart effort; however, no set date has been established. This is an open item.

Software Configuration Control

The Certification Plan for Reactor Analysis Computer Codes includes a configuration control action matrix. The matrix addresses the following required items relating to software configuration control:

- o code proprietors
- o proprietor's code log book
- o user-manual availability
- o training package
- o code source listing location
- o test problem and results location
- o control tracking system
- o system dependent code documentation
- o code interaction impact
- o range of applicability
- o backup versions
- o shell scripts and banners
- o configuration control review
- o configuration control in place (management reporting of completion)

These items comply with the restart criteria. In lieu of full configuration control, the exemption requests are needed to address short-term deviations and compensatory actions for configuration control. This is an open item.

Documentation

The V&V plan provides a record of the V&V plan action matrix for each code. This plan, which is applicable to the SRS organization, calls for completion of the following sections: "Basics," "Verification of Theory," "Validation with Experiments," "Validation by Benchmarks," and "Conclusion." Successful completion of this V&V plan, along with a functioning QA recordkeeping system, will ensure that user documentation is complete. The V&V action matrix includes the required documentation as follows:

- o Basics - manuals, configuration control plan, and code portability
- o Verification of Theory - appropriate theory, theory documentation, coding consistency, theory verified conceptually, theory verified by experiments, and theory documentation adequacy
- o Validation with Experiments - tests in experimental facilities, tests in operating facilities, data from operating facilities, test data documentation, appropriate data quality, validation performance, and validation documentation adequacy

- o Validation by Benchmarks - benchmark requirements identification, similar code comparison, exact solution comparison, industry benchmark comparison, comparisons documentation, and benchmark documentation adequacy
- o Conclusion - verification review, verification completion, standard sets of test problems, validation/benchmarking review, and validation/benchmarking completion

Problem Reporting and Corrective Action

The staff's review indicated that the next revision of the plan should address the control of code output, which would ensure full retrieval of safety-related analyses that might be affected by any software errors or failures during the software life cycle. WSRC agrees that the control of code output is an important aspect of safety. However, WSRC feels that the existing Nuclear Reactor Technology and Scientific Computations (NRTSC) QA procedures (Reference 12) are adequate and that no specific actions need be described in the certification plan.

The staff examined the procedures provided by WSRC staff and concluded that the control of code output may have some deficiencies. The procedures address the control of the analysis reports and may not fully address the control of code output. Furthermore, the staff determined that, site-wide, the control of code output is inconsistently practiced (since no formal procedure exists) and that there is no facility to provide controlled/retrievable QA record storage for code output, which would enable effective problem reporting and corrective action for computer generated critical calculations. This is an open item.

Access Control

The Certification Plan for Reactor Analysis Computer Codes includes a user qualification section. This section contains a user qualification and function checklist that specifies the minimum requirements for each user level and the associated access (authorization or control) level (i.e., from apprentice to proprietor) necessary to perform required code alterations. DOE review of this Certification Plan for Reactor Analysis Computer Codes indicated that it is an acceptable plan for access control. At present, master copies for many of the codes listed in the Certification Plan are archived in the Software Configuration Management System (SCMS), which is controlled by the NRTSC Scientific Computations Section (SCS).

Software Procurement

For those codes in the November 15, 1989, V&V plan that were acquired from outside software sources, successful completion of this plan will satisfy the long-term criteria for software procurement.

An exemption request documenting both the short-term deviation from the V&V requirements and the necessary justification with compensatory actions was submitted by NRTSC for approval on May 23, 1990. The request addresses two offsite codes: TRAC and RELAP, and has been approved by DOE.

SRSP0 had a concern on the adequacy of site-wide software procurement management and quality assurance in early 1989. This was related to the technical support by the Engineering Project Division necessary for the seismic analysis program (SER Section 6.1). The SRL NRTSC has been working to resolve this issue. A draft engineering software management strategy and principle was provided to DOE. The completed document is expected for final review prior to restart. This is an open item.

Records

The V&V plan provides a record of the V&V plan action matrix for each listed code. Successful completion of this V&V plan, along with a functioning QA recordkeeping system, will satisfy the restart criteria for records. The adequacy of the implementation of the QA recordkeeping system will be covered in the implementation review.

Certification

The Certification Plan for Reactor Analysis Computer Codes contains a "user qualification's" section that addresses the following items:

- o User Level Description (The user-level designation of an individual is the joint responsibility of the task leader and the appropriate management.)
- o The Apprentice User
- o The Cognizant User
- o The Code Proprietor
- o Minimum Requirements for User Levels
- o User Qualification and Function Checklist
- o Additional Training (special training seminars on the computer code's numerical methods and practical use)

These items are consistent with the restart criteria for certification. The staff's review indicates that the plan should address the establishment of a separate "User Guides/Information File" for collecting and assimilating experience in application of safety-related software. In Reference 8, WSRC agreed that the implementation of such a database would be a supplement to the code user's manual. This database will help in strengthening the user's competency in applying the codes, and user's training and qualification are essential parts of the certification process.

5.8.4 Conclusion

Based on the above evaluation, the staff concludes that acceptable progress has been made toward meeting the restart criteria for software quality assurance. Open items are clearly identified.

The staff will update its evaluation and document the results in a supplement to the SER.

5.8.6 References

1. WSRC-TR-90-397, "Task Assessment Schedule for Certification of Reactor Analysis Computer Codes," September 30, 1990.
2. American National Standards Institutes/American Society of Mechanical Engineers, ANSI/ASME Standard NQA-2, part 2.7 (draft, June 14, 1989).
3. National Academy of Sciences Report, 1987.
4. WSRC, WSRC-RP-89-368, "WSRC Reactor Operations Management Plan (ROMP)," February 28, 1990.
5. WSRC, "Reactor Restart SQA Assessment."
6. WSRC, Letter from P. W. Dickson to R. E. Tiller, "Verification and Validation of Reactor Analysis Computer Codes," dated November 14, 1989.
7. SRL/NRTSC, "Certification Plan for Reactor Analysis Computer Codes," dated January 15, 1990.
8. WSRC-OPS-RRP-900057, WSRC Letter from P. Dickson to F. McCoy, May 14, 1990.
9. SRSP0 Letter from F. McCoy to J. Gallagher, July 17, 1990.
10. SRL/SARM, WSRC-RP-90-433, "Verification and Validation Plan for Safety and PRA Codes" (draft), May 1990.
11. SRL/SARM, WSRC-TR-90-183, "Certification Plan for Safety and PRA Codes" (draft), May 1990.
12. WSRC, DPSTM-88-700-9, "QA Procedures for the Nuclear Reactor Technology and Scientific Computations Program Management Team," August 1988 (including the following sections: QAP-II-7, "Documentation of Critical Calculations"; QAP-II-11, "QA Records"; QAP-II-12, "Laboratory Notebooks," QAP-II-13, "Reporting of Results and Release of Preliminary Data"; QAP-II-14, "Technical Reviews"; and QAP-IV-10, "Software Production Runs").

6 DESIGN ISSUES

6.1 Seismic

6.1.1 Background

The seismic section of the Safety Evaluation Report (SER) has been divided into seven subsections: Geology, Seismic Ground Motion, Geotechnical Engineering and Site Soil Conditions, System Response to Seismic Motion, Structures and Foundations, Piping, and Equipment.

The "Savannah River Plant K-Reactor Restart Strategy" (also known as the Charlotte Criteria) issued in November 1988 requires WSRC to demonstrate adequate seismic resistance. This requirement states that the reactor will "provide the capability to shutdown, cool, and monitor core status for a design basis earthquake (0.2g) for a period of 72 hours after the event." Implicit in this requirement is that the pressure boundary of the Process Water System (PWS) remains intact. Core shutdown is secured with a soluble poison, the loss or dilution of which cannot be tolerated. It should be noted that, while the Moderator Recovery System (MRS) can recover and recirculate poisoned primary coolant, this coolant should not be diluted. In order to accomplish this, the following systems must be seismically qualified: the Supplementary Safety System (SSS), the PWS, the Cooling Water System (CWS), the Diesel Power System (DPS), and those systems that provide the core monitoring capability. In addition, as a result of the Westinghouse Independent Safety Review (WISR), the MRS and the Cooling Water Removal and Disposal System (CWRDS) were added to the list of systems that require seismic qualification prior to restart. In order to demonstrate seismic qualification, analyses and approved seismic walkdowns were used.

Additional testing and analysis are planned as a result of this evaluation. Open items have been identified and described in each section of the seismic portion of the SER. These open items will continue to be examined and will be closed out as part of the evaluation process.

6.1.2 Restart Criteria

Many of the restart criteria for the seismic evaluation are based on Sections 2.5 and 3.7 of the U. S. Nuclear Regulatory Commission's (NRC's) Standard Review Plan (NUREG-0800). A detailed listing of the restart criteria can be found as Attachment A, with reference to the specific Standard Review Plan section, as appropriate.

6.1.3 Geology

This subsection of the Savannah River Site (SRS) SER contains an evaluation of material that describes the geologic conditions at the SRS. In addition to providing support to the development of the seismic hazard and the design basis earthquake, the geologic descriptions are required to identify any characteristics of the overburden soil that could result in undesirable foundation behavior, including foundation behavior during a seismic event.

Evaluation

The DOE staff found that the data in the SRS Safety Analysis Report (SAR) and other reviewed seismic-related data and references adequately describe the regional and site stratigraphy, the geologic history of the region, and the structural geology of the site. Data related to determining the age of last movement for faults in the region have been reviewed, and all faults are judged to be non-capable. As discussed below, field investigations are underway to further reduce the uncertainty regarding the age of last movement for the Pen Branch Fault. Current information suggests that the Pen Branch is non-capable.

Site Geology

The staff has reviewed the material of record that forms the bases for the interpretation of geologic data at SRP. The data provided in the material of record are consistent with generally accepted geologic and structural models of the area (Cook, 1936; Geological Society of America Bulletin, 1972; Marine, et al., 1974).

The SRS reactor sites are located on the Upper Atlantic Coastal Plain. The Atlantic Coastal Plain extends from the Fall Line of the Piedmont to the Continental Shelf. The Coastal Plain is underlain with a seaward dipping wedge of unconsolidated to semi-consolidated sediments that range in thickness from 0 feet at the Fall Line to 4000 feet at the South Carolina coast. The wedge thickness at the plant is about 1000 feet. These sediments have been created by the deposition of material from both marine and non-marine sources. Within the SRS site boundaries, the formations below this wedge of sediment range from Crystalline, Paleozoic, and older rocks that represent the SE dipping Piedmont to predominantly Mesozoic rocks within the Dunbarton Basin. Data available from boreholes and wells in the area provide the bases for identifying the geologic material and structural relationship in this sediment wedge. During the construction period, soil data were collected (Du Pont, 1952) at each of the reactor sites. These data were collected from boreholes drilled down to about 200 feet in depth.

Two seismic reflection studies have been conducted at the site since the 1970s. The first (D'Appolonia, 1978) covered 13,000 feet aligned along 4 lines located 5 miles from K-Reactor. The second survey was done by CONOCO during the time period of 1987 through 1989. The CONOCO seismic surveys confirmed that the location of the basement is about 1000 feet below the surface and transitions from Crystalline material in the north into a Triassic Basin towards the southeast and that the different younger soil layers contained within the overburden material are of fairly uniform stiffness. The results of these surveys were reviewed by DOE staff and found to be based on sound field work with reasonable interpretations of the geologic and structural data. Reprocessing of the CONOCO seismic data to better delineate shallow geologic features will proceed over the next year. This reprocessing will aid in reducing uncertainty in the age of last movement for faults such as the Pen Branch fault, which is discussed in more detail below.

The characterization of the soil overburden at SRS as a fairly uniform sand to sand/clay layer of about 1000 feet thickness and having a shear wave velocity slightly larger than 1000 fps is consistent with the data collected at the site. This description is also consistent with the geologic history of the area. As discussed in Section 6.1.5, a soils characterization program has been initiated to further document the near surface material properties at the reactor site.

Studies of the faulting in the site region have been completed by a number of investigators, for example, Higgins (1978), Cramer and Arden (1978), Wentworth and Mergner-Keefer (1981), and Faye and Prowell (1982), and summarized by URS/Blume (1982) in the SRS SAR as well as in the Final Safety Analysis Report (FSAR) for the Vogtle nuclear plant. Several faults have been postulated to offset the Coastal Plain material, including the Belair fault to the north of the site and the Millett fault and Statesboro faults at the south end and south of the site. At various times, these faults have been studied using techniques such as geologic mapping of surface units and correlation of subsurface units from cored boreholes and geophysical techniques, such as seismic reflection techniques. Results of these studies indicate that material above the Triassic/Cretaceous contact is continuous; thus, these faults are not capable and do not present a seismic hazard to SRS.

As reported in the SRS SAR, the Dunbarton Triassic basin is located beneath SRS. The boundary between the Triassic sediments and the Jurassic basement form faults. Additionally, faults have been identified within the Dunbarton Basin (see SRS SAR Figure 2.5-25). Data collected from several wells drilled into the Triassic sediments and from seismic reflection surveys indicate displacements across many of the faults occurred prior to the deposition of the Coastal Plain sediments.

Recent work (Snipes, et al., 1989) presents data describing a newly discovered fault, the Pen Branch Fault. Drilling data suggest that this fault was active in post-Eocene time. There is no evidence of any discontinuity in the younger Quaternary formations, although such material is not pervasive throughout the site. There are no surface drainage features or geomorphic features that would be indicative of recent activity along the fault. Evidence suggests that the fault is not capable. This conclusion is consistent with conclusions reached by the NRC in a recent evaluation for the Vogtle site. To address the remaining uncertainty regarding the age of last movement, field investigations are underway to confirm that the Pen Branch Fault is not a capable fault. Investigations planned include high resolution reflection profiles, reinterpretation of the Conoco reflection profiles, and surface geologic studies.

The SRS SAR also includes a discussion of the engineering significance of the regional and site geology, including the fact that dissolution of calcareous material at a depth of about 120 to 130 feet is a potential geologic hazard that must be considered in the development of the site. The soil overburden contains calcareous material, indicating that dissolution of the material and the potential for cavities in the soil are aspects of soil behavior that must be considered at SRS. This problem was investigated during the site studies conducted prior to construction and resulted in the use of grout in the

reactor construction areas. Similar grouting efforts have not proceeded in other areas, such as near the Cooling Water Basin, although a number of holes were drilled in this area. A soils program is underway to fully evaluate the success of past grouting efforts and to characterize other areas near the reactor site to determine the presence and extent of any soft areas or dissolution areas (see also Section 6.1.5).

Open Items

With the exception of the following open items, the restart criteria have been satisfied.

- (1) A reactor site-specific soils program to evaluate the success of past grouting efforts and to characterize the dynamic properties of the underlying soils must be performed prior to restart (also see Section 6.1.5).
- (2) Field investigation on the SRS site must be continued as part of the effort to further confirm that the Pen Branch Fault is not a capable fault.

6.1.4 Seismic Ground Motion

This subsection of the SRS SER contains an evaluation of the definition of the seismic ground motion at the SRS used to define the design basis earthquake (DBE) and includes a discussion of the following: (1) seismicity in the area, (2) geologic and tectonic characteristics of the site and region, (3) correlation of earthquake activity with geologic structure or tectonic provinces, (4) maximum earthquake potential, (5) seismic-wave transmission characteristics of the site, and (6) determination of the DBE.

Evaluation

Seismicity

In 1982, URS/Blume performed a systematic evaluation of the seismicity of the SRS reactors and compiled a catalog of preinstrumental data for South Carolina covering the period of 1698-1979 and including the Charleston 1886 event. URS/Blume also generated a catalog of instrumental data for South Carolina covering the period of 1945-1979, gathering published information regarding earthquake intensity recurrence pertinent to the region and examining this information for applicability to SRS.

The SRS SAR also cites recent work by Bollinger et al. (1987) on the seismicity of the southeastern United States, which includes data for the period of 1698-1986. Based on this work, the SAR presents interpretations of seismicity in terms of focal depths and mechanisms. Both the URS/Blume study and the SAR conclude that the seismicity of interest to SRS is dominated by the Charleston 1886 earthquake (estimated to have produced an MM intensity of VI at the site). In addition to the material in the SRS SAR, the Southeastern U.S. Seismic Network Bulletins and the Vogtle FSAR were reviewed to gain an understanding of the seismicity pattern throughout the southeastern United

States. The description of the regional seismicity provided by these studies satisfies the restart criteria.

Geologic and Tectonic Characteristics

The major geologic and tectonic provinces in the region of SRS are associated with those of the southeastern United States and are widely discussed in the available literature. The major provinces of interest with regard to SRS are the Blue Ridge (northwest), Piedmont (central), and Upper Coastal Plain (southwest). The Valley and Ridge province (part of the Appalachian Mountains outcropping) represents a fourth province, which is located at the northwest boundary of the region. These provinces are northeast trending, with the Coastal Plain east of the Fall Line and the Piedmont and Blue Ridge west of it.

The SRS is located in the Upper Coastal Plain, approximately 25 to 30 miles southeast of the Fall Line. The Upper Coastal Plain consists of a wedge of unconsolidated to semi-consolidated sediments overlying a bedrock formed during the Pre-Cambrian and Paleozoic eras. The thickness of the sediment layer at the SRS is about 900-1000 feet.

The DOE staff concludes that the overall interpretation of the regional geologic and tectonic characteristics of the SRS region given in the SRS SAR is consistent with available information from geological and geophysical investigations of the region.

Correlation of Earthquake Activity with Geologic Structure of Tectonic Provinces

The DOE staff believes that on the basis of the information currently available, it is not possible to demonstrate a definite correlation between the regional seismicity and specific geologic structures. Although available geological and geophysical interpretations generally seem to offer reasonable explanations, no definite conclusions can be drawn on the cause of the regional seismicity by identifying specific seismogenic sources. The DOE staff also believes that intraplate seismicity will continue to be a subject of future investigation and debate.

Two approaches can be used to define vibratory ground motion, given the lack of correlation between seismicity and geologic structure. These approaches have been labeled the deterministic approach (that based on concepts contained within the Code of Federal Regulations [CFR], Title 10, Part 100, Appendix A) and the probabilistic seismic hazard approach. In the deterministic approach, specific assumptions are made in terms of earthquake size and distance from the site. For the SRS, a determination needs to be made regarding the closest approach of an event similar to the 1886 Charleston earthquake. In reviewing recent information, including information compiled for the Lawrence Livermore National Laboratory (LLNL) and Electric Power Research Institute (EPRI) probabilistic seismic hazard studies and studies related to paleoseismic evidence of multiple earthquakes near Charleston (Talwani and Cox, 1985; Obermeier et al., 1986; and Weems et al., 1986), it appears as if the majority of evidence and scientific opinion suggest that the Charleston earthquake was

associated with a localized seismic source zone. A recent paper by the NRC staff states that "the assumption in past licensing decisions of confining the recurrence of the 1886 Charleston, South Carolina to the Charleston region has not resulted in inappropriately high levels of seismic hazard at existing nuclear power plants" (Giese-Koch, et al., 1990).

Consistent with the Standard Review Plan Section 2.5.2.4, the closest distance of potential seismic sources should be determined. For the deterministic estimates of ground motion, it is assumed that the Charleston event re-occurs near Charleston, or about 120 kilometers from the SRS. The largest historical event in the remainder of the Coastal Plain is intensity = VII, and this event is assumed to occur within 0 to 25 kilometers of the site at about 15 kilometers.

It is further expected that low magnitude earthquakes such as the 1985 and 1988 Savannah River Plant earthquakes are likely to reoccur as a result of local seismogenic sources. A better understanding of the cause of lower level seismicity at SRS may emerge from future confirmatory investigations related to the Pen Branch Fault. Such investigations are expected to provide an age estimate of the last displacement of the fault using techniques such as surface mapping, core borings, and seismic reflection methods.

Design Peak Ground Acceleration (PGA)

The DBE ground motion that has been used to upgrade the K-Reactor is characterized by a peak horizontal acceleration of 0.20g associated with a Regulatory Guide 1.60 response spectra. At this time site-specific deterministic estimates of ground motion have not been completed. The DOE staff made estimates of ground motion for the Coastal Plain and Charleston sources and found that a Regulatory Guide 1.60 spectra anchored to a peak horizontal peak acceleration of 0.20g, is appropriate for the engineering seismic analysis being completed. WSRC is completing a confirmatory assessment of the DBE ground motion that will include site-specific deterministic estimates of ground motion for both the Coastal Plain source and the Charleston source, consistent with Standard Review Plan Section 2.5.2. This evaluation will include an assessment of the appropriate magnitude, distance, and method of estimating ground motion. The results of the WSRC work will be compared to the DBE response spectra.

DOE has also considered probabilistic seismic hazard curves developed for the SRS region to evaluate the adequacy of the PGA and response spectra. Probabilistic seismic hazard estimates allow for the incorporation of uncertainties and alternative hypotheses on many of the hazard input parameters including seismic source zonation, earthquake occurrence rates, upper magnitude cut off, and ground motion attenuation. A comparison of recent studies is presented as Table 6.1. The first three rows of the table summarize PGAs at three levels of annual probability of exceedance (10^{-3} , 2×10^{-4} , and 10^{-4} respectively). The bottom row in the table shows corresponding estimates of the annual exceedance probabilities for PGA = 0.20 g. The values compared in the table are: (1) URS/Blume mean value for SRS, (2) LLNL's 1984 best estimates, (3) LLNL's 1989 50 percent, 85 percent values and mean values, and (4) EPRI's 1989 50 percent, 85 percent, and mean values.

Table 6.1
COMPARATIVE RESULTS FROM SEISMIC HAZARD STUDIES

PROB. OF EXCEED	URS/ BLUME MEAN	LLNL 1984 BEST	LLNL 1989 50%	85%	MEAN	EPRI 1989 50%	85%	MEAN
0.001	0.12	0.11	<.10	0.20	0.26	<.10	<.10	<.10
0.0002	0.20	0.18	0.15	0.38	0.52	0.10	0.18	0.15
0.0001	0.24	0.24	0.20	0.48	0.65	0.15	0.24	0.20
ANNUAL PROBABILITY OF EXCEEDANCE								
0.2g	.0002	.00018	.0001	.001	.002	.00004	.00016	.0001

The probabilistic results suggest that the 0.20g DBE peak acceleration has a return period on the order of 1,000 to 10,000 years. These return periods are consistent with return periods for nuclear power plants. The uniform hazard spectra (UHS) for SRS generated by LLNL (1989) and EPRI (1989) have also been reviewed. Comparisons of the 1,000 year and 10,000 year UHS also support the conclusion that the Regulatory Guide 1.60 spectra shape is adequate for restart. The DOE staff recognizes the uncertainty associated with probabilistic seismic hazard estimates and notes that these uncertainties were also recognized by the NRC. DOE continues to follow the guidance provided by the NRC in terms of using probabilistic seismic hazard results to assess facility design.

Design Ground Response Spectra

Upon reviewing previous design spectra used at Savannah River, the DOE staff concluded that the spectral shape of Regulatory Guide 1.60 should be considered for SRS restart in terms of engineering assessment of structures, systems, and components. The Regulatory Guide 1.60 spectra envelop the previously considered URS/Blume spectra, are generally accepted to be sufficiently broad-banded, and contain adequate amplification over the frequency range of interest for SRS. These spectra were recently used for performing soil-structure interaction and component evaluations for SRS and, as discussed above, are consistent with recent estimates of ground motion for the SRS. The deterministic calculation of ground motion being completed will be compared to the Regulatory Guide 1.60 spectral shape.

Seismic Wave Transmission Characteristics of the Site

Information regarding the seismic wave transmission at SRS in terms of frequencies of interest is uncertain due to a lack of recent in-situ property soil data from the K-Reactor site vicinity. Therefore, possible modifications of the surface free-field motion due to site response characteristics have not been determined. WSRC is calculating site-specific estimates of ground motion

which will be considered the local site-specific soil properties (also see Section 6.1.5).

Open Items

At this time there are no open items in this area; however, the following confirmatory issue has been identified.

- o Development of site-specific deterministic estimates of ground motion to confirm the appropriateness of the DBE.

6.1.5 Geotechnical Engineering and Site Soil Conditions

This subsection contains an evaluation of the impact of the foundation soil properties on the response of the K-Reactor building facility when it is subjected to design static and dynamic loadings. The topics discussed in this section include an evaluation of: (1) the potential behavior of the foundation soils to the applied dead and live loads imposed by the building foundations; (2) the potential impact of long-term solution processes on the development of voids, soft zones, or sinks in the deeper foundation soils on the behavior of the facility for both long-term static loadings and seismic inputs; (3) the potential for the development of ground failure from soil settlement or liquefaction of the foundation soils when subjected to the design seismic loadings; and (4) the determination of the variability of the dynamic soil properties of the foundation soils used for soil-structure interaction analyses.

Evaluation

Bearing Capacity

Seven borings were taken in the vicinity of the K-Reactor building during the original construction phase. The logs for three borings were available for inspection. Soil descriptions and blow count data from Standard Penetration Tests (SPTs) obtained from these borings indicate that the soils are relatively dense sands and silts, having SPT blow counts of about 25 or more. This information corroborates data obtained from borings taken at the L-area, which exhibited similar profiles. Estimates of foundation bearing pressures for the K-Reactor stack building obtained from the design dead, live, and seismic loadings reach a maximum of 8.6 ksf. This is significantly less than the ultimate bearing capacity of the foundation, which was estimated to be well over 200 ksf. Under the reactor building, foundation pressures were estimated to be 8.2 ksf, while the ultimate bearing capacity of the foundation soils was estimated to be over 500 ksf. Visual inspection of the facility indicates that no significant deleterious effects of structural movement due to foundation settlement has occurred over the life of the building, supporting this judgement.

Liquefaction Potential and Evaluation of Soil Settlement

In the K-Reactor area, the highest ground water level is 35 feet below the ground surface. Therefore, only the soils below the 35-foot depth are of

interest. The soil descriptions from boring K-5 within a depth of 40 feet are indicated to be clayey sands; that is, the soil has a significant cohesive binder. Soils subject to liquefaction are cohesionless sands and silts. Finally, the SPT blow counts to a depth of 50 feet are indicated to be well above 30 blows per foot from the original soil investigation made by the Corps of Engineers in 1951. Soils subject to liquefaction typically have blow counts of less than 30 to a depth of 50 feet. Thus, the K-Reactor soils do not have the general characteristics which indicate the potential for liquefaction down to a depth of at least 50 feet.

During the original site investigations, it was found that the site geology indicates the potential for dissolution of the calcareous material contained in the McBean soils located at depths from 80 feet to 140 feet. Local soft spots were found during the original drilling conducted prior to construction, which led to the use of grout to stabilize these soils. At the K-Reactor building, 289 borings were drilled into the McBean soils on a 50-foot staggered drill pattern, covering an area extending on all sides about 50 feet past the building footprint. Of the total number of borings placed, 76 were found to have soft spots indicating the need for grout. These grouted holes were clustered in four primary areas around the building foundation. About 5,000 cubic yards of grout were injected into the McBean soils to stabilize these soft spots.

Following the grout program, an additional 23 test borings were drilled in the K-Reactor building area to attempt to assess the success of the grout program. Of these borings, 20 were placed in areas where significant quantities of grout had been injected. Only 1 of these 20 holes showed any loss of water during the drilling operation; the hole was later grouted.

To evaluate the grout program, a simple calculation was performed in which the amount of grout required to change an estimated volume of loose soil to a medium dense condition was calculated. Using the results of the drilling program as a guide, approximately 13,600 cubic yards of loose soil were estimated under the building footprint within the McBean soils. To change this soil to a satisfactory density condition, it was estimated that about 0.25 cubic yard/cubic yard of total volume of the soil is required, leading to a required grout volume of 3400 cubic yards. Since about 5000 cubic yards of grout were actually pumped under the building, it can reasonably be estimated that any existing soft spots were suitably filled.

This simple calculation supports the conclusion that can be reached from the results of the grout test program itself. The available evidence indicates that sufficient grout was placed to fill any significant voids or soft spots under the K-Reactor building and ensures that the soil in the affected zone is sufficiently stiff to eliminate any concerns with the future response of the soils under the building. There remains uncertainty, however, regarding the potential for dissolution near the Cooling Water Basin. WSRC has initiated a soil boring program to address this issue. The soils data will be used to assess the potential for seismically induced soil settlement given the occurrence of a DBE.

Dynamic Soil Properties for Use in SSI Analyses

Current information on dynamic soil properties indicates that the soil properties used in the soil structure interaction (SSI) analyses for the K-Reactor building were estimated from the SPT blow count data from boring K-5. These data, in turn, were correlated with low-strain, shear wave velocity data for sand, which was obtained from the open literature. A comparison of the data with cross-hole measurements obtained from other areas of SRS indicates that, on the average, the low strain data estimates are reasonable. As described in Section 8.4.1 of this SER, the latest SASSI calculations, which include soil layering effects, indicate that the response spectra calculated using the uniform property data are conservative.

D'Appolonia (1979) also performed a cross-hole survey in three borings at area 200-B. These borings extended to depths of 300 feet. The results of the cross-hole tests indicate fairly uniform shear wave velocities of slightly higher than 1000 feet per second (fps). Down-hole tests were performed in the same holes and confirmed the results of the cross-hole tests.

While current assumptions appear to be reasonable, additional soils data are being collected around the K-Reactor and Cooling Water Basin to provide a more complete in-situ property data base to confirm that soil property assumptions made for the soil structure interaction calculations are appropriate.

Open Items

On the basis of the results of this evaluation, the DOE staff concludes that the geotechnical engineering evaluations satisfy the restart criteria for SRS restart except for the following open item.

- o A soils characterization program is being developed for the K-Reactor and Cooling Water Basin site to support the current seismic analyses of structures. The program scope includes cone penetrometer tests, down-hole and cross-hole tests, laboratory testing of undisturbed soil samples, and grout-take tests (if required).

6.1.6 System Response to Seismic Motions

This subsection addresses the evaluation of the models and methods used to develop the floor response spectra (FRS) used for piping and equipment seismic qualification and seismic loads acting on structures. All of the systems required for restart are contained in the reactor building, the stack building, the cooling water reservoir (186 Basin), and the process effluent sump structure.

Evaluation

To determine the adequacy of the K-Reactor building for restart, the DOE staff reviewed in detail documents generated by URS/Blume (August 1989, December 1989), BNL (1989), and Quadrex Corporation (1987) and compared these documents with the restart criteria.

The synthetic time history used for the analysis has response spectra that envelop the criteria response spectra specified for the site. The SSI models have been shown to adequately include the effects of frequency dependence, soil layering, foundation embedment, depth to bedrock, strain dependence of soil properties, torsion effects, and building-to-building interaction. The structural models satisfy the restart criteria.

Conformance with the recommendations of Regulatory Guides 1.60 and 1.61 ensures that the seismic inputs for evaluation of the K-Reactor structures are adequately defined. This forms a conservative basis for these structures, piping, and equipment to withstand the seismic loadings.

On the basis of this evaluation, the staff concludes that the methods used to generate the seismic loads and FRS for the K-Reactor structures are acceptable and that the restart criteria for SRS restart are satisfied with the exception of the remaining open item discussed at the end of this section.

Reactor Building

The analysis of record that implements the restart criteria for the reactor building was performed by URS/Blume (August 1989). DOE staff comments and responses from SRS regarding the extent to which the analysis methodologies capture the criteria requirements were documented. The primary thrust of these discussions centered around the extent to which the SSI model adequately represents the effects of embedment, soil layering, high frequencies, and torsional effects.

The analysis of the K-Reactor building was based on the FREDA computer code. This solution used a lumped mass stick model of the structure connected to the free-field through SSI frequency dependent functions. The basic model did not include vertical and torsional responses. The SSI coefficients used in the basic model satisfy the restart criterion except for the consideration of layering and high-frequency effects. The staff concludes that free-field input motion generated by URS/Blume and the damping values used satisfy the restart criteria.

In response to the comments raised by DOE staff concerning high frequency variations, layering effects, and torsional effects, URS/Blume performed two additional computations. High frequency variations and layering effects were treated in an analysis using the SASSI computer code. Torsional effects were included in an additional FREDA analysis. Spectra generated from these results were compared with the first FREDA spectra. It is concluded that the original spectra developed from FREDA for elevation 34 feet and lower are conservative. At the higher elevations, it is recommended that SASSI spectra be amplified by torsion/no-torsion ratios and used for equipment and piping analyses.

Based on a preliminary evaluation, the staff concludes that taking the vertical FRS equal to the horizontal FRS is conservative. The staff's evaluation included a comparison of the vertical/horizontal ratios of free field motion, SSI parameters, and structural amplification. The comparison of structural amplification is based on a review of the structural drawings that

indicate that the most flexible slab in the reactor building is a 2-foot-thick slab spanning 34 feet. This remains an open item, pending review of additional analysis by WSRC to verify that the slab discussed above is the bounding condition (i.e., most flexible).

Stack Building

URS/Blume (December 1989) also performed the seismic analysis of the stack building, which consisted of the following three analyses.

- o Phase I Analysis -- In this phase, a three-dimensional model was used to perform parametric studies with the objective of identifying required fixes to the structure. The major modifications proposed were the addition of three, 5-foot-thick shear walls between four the main columns and the strengthening of one of the walls in the fan room. This analysis is deficient in that it does not include the effects of building-to-building interaction (between the reactor building and stack building), soil layering, and foundation flexibility.
- o Phase II Analysis -- This phase resolves the three deficiencies of the Phase I analysis. There are two difficulties with the Phase II stack building analyses. First, the stress analysis indicates several areas of distress. WSRC designed structural modifications to alleviate the overstress conditions. These design modifications are now being implemented. Second, when some of the stack building modifications were being made, it was observed that the as-built stack building differed slightly from the drawings.
- o Phase III Analysis -- URS/Blume developed a Phase III model that incorporated the design changes and as-built conditions. The Phase III analysis satisfies all of the restart criteria.

Cooling Water Reservoir

Quadrex Corporation performed the seismic analyses for the cooling water reservoir (CWR). Seismic analyses of these structures were performed using finite element representations of the soil and water and beam element representations of the reinforced concrete walls. Three two-dimensional models representing vertical sections taken through the structure were analyzed. Two deficiencies were found with these analyses. First, the 4 percent damped Blume seismic input was used rather than the 7 percent damped input from Regulatory Guide 1.60. Second, no variations in soil properties were considered.

The staff performed a confirmatory analysis addressing these two concerns and concluded that the Quadrex analysis was adequate. In view of the low safety factors, however, the Defense Nuclear Facilities Safety Board (DNFSB) concluded that a more complete set of analyses should be performed.

WSRC has since completed the analyses of the CWR structure (WSRC-RP-90-1338, December, 1990) for seismic response using the Regulatory Guide 1.60 free-field motion anchored at 0.2g. Soil structure interaction and hydrodynamic effects were included in the two-dimensional SuperFLUSH models used to obtain seismic forces and bending moments and acceleration floor response spectra. Variation in soil properties were considered by using best estimate, 2 times best estimate, and 1/2 best estimate low strain shear modulus values in the analytical models. Three-dimensional static models were developed to account for three-dimensional effects and to evaluate stiffness coefficients for the two-dimensional models. The DOE staff concluded that the analytical modeling techniques, the seismic input, and the treatment of soil properties satisfy the restart criteria.

Process Effluent Sump Structure

Quadrex used the same procedures to analyze the process effluent sump (PES) structure as were used for the cooling water reservoir. Plane sections were taken through the structure in the E-W and N-S directions. The same review comments apply to this structure as to the cooling water reservoir.

WSRC has also completed an analyses of the PES structure (WSRC-RP-90-1338, December, 1990) for seismic response using the RG 1.60 free-field motion anchored at 0.2g. Similar to the analysis of the CWR, SuperFLUSH models were used to obtain seismic response and acceleration floor response spectra. Variation in soil properties were considered by using best estimate, 2 times best estimate, and 1/2 best estimate low strain shear modulus values in the analytical models. Since the PES structure is box-like, three-dimensional static models were developed to account for the three-dimensional distribution of loading. Forces and moments were evaluated by inputting seismic displacements obtained from the SuperFLUSH 2-d models into the three-dimensional models. DOE Staff concluded that the analytical modeling techniques, the seismic input, and the treatment of soil properties used for the PES structural seismic analysis adequately satisfy the criteria established for restart.

Open Items

The following open item requires resolution prior to restart.

- o It must be demonstrated that the floor frequencies of the reactor building are sufficiently high to ensure that structural amplifications of the vertical motion are no larger than structural amplifications of the horizontal motion. This is required if the vertical FRS is to be taken as 2/3 of the horizontal FRS. The staff performed an analysis of a floor that appears to be the most critical and found that the floor frequency is sufficiently high. It remains to be confirmed that other floors are not more flexible than the floor selected for the analysis.

Concerning the remaining open issue listed above, WSRC has completed an analysis of the K-Reactor building floor frequencies (WSRC-RP-90-1328) and the DOE staff is in the process of reviewing this work.

6.1.7 Structures and Foundations

This subsection contains an evaluation of: (1) the seismic-induced loads acting on the SRS structures (i.e., the K-Reactor building, the stack building, the CWR structure, and the PES structure); (2) the combination of these loads with other loads; and (3) the structural stresses.

Evaluation

To determine the adequacy of the K-Reactor for restart, the staff reviewed documents by Du Pont (1969), URS/Blume (1977, 1978, August 1989, December 1989), and Quadrex Corporation (1987). These documents were compared with the restart criteria.

With the exception of the open item discussed at the end of this section, the strength evaluation of the structures and foundations contained in the material of record satisfies the restart criteria.

Reactor Building

The first evaluation of the reactor building seismic response that included dynamic effects was done in 1969 by Du Pont. A lumped mass stick model of the structure was developed for evaluation of the building's seismic response. The basemat motion was specified as that associated with the Housner seismic criteria. The only problems found were overstressing in the walls at the base of the actuator tower and the girders supporting the actuator tower.

In 1977, URS/Blume made a seismic study of the reactor building using the Regulatory Guide 1.60 free-field motion as input anchored at 0.2g. URS/Blume found that overstress conditions in the actuator tower were about twice as severe as for the Housner free-field input. URS/Blume also found that buttresses, spanning from the tower walls to the main walls of the reactor building, would relieve the overstress conditions in the actuator tower and the supporting girders. These buttresses have since been added to the actuator towers for all the SRS reactors.

The latest structural evaluation of the K-Reactor building was completed in August 1989 by URS/Blume. The URS/Blume seismic analysis resulted in a time history of shear force at each elevation in the structure for each of the two directions of seismic input. URS/Blume found the critical shear stress to be 123 psi, as compared with an allowable stress of 188 psi. An evaluation of these analyses indicates that the shear wall stress comparisons satisfy the restart criteria.

There are no stress evaluations of floor or foundation slabs available. Seismic effects usually have small impact on stresses in floor slabs. If the slab is adequate for the dead and live load case, the slab is probably

adequate when seismic effects are included and seismic allowable stresses used.

A foundation bearing pressure computation against which the second restart criterion can be tested is not available for SRS. A calculation performed by DOE staff determined that the maximum soil bearing pressure during an earthquake would be 8.24 ksf. The value is well below the design bearing capacity which has been shown to be in excess of 500 ksf.

Stack Building

The first serious examination of the seismic response of the stack building was performed in 1969 by Du Pont. This evaluation indicated that the base of the stack was overstressed when the Housner free-field input was used.

In a 1977 URS/Blume study, which considered the effect of using the Regulatory Guide 1.60 free-field input, even more severe overstress in the stack was noted. The stack response to the RG 1.60 input was again considered in 1978 when some inelastic deformation was permitted. Based on this analysis, the wall thickness of the bottom third of the stack was increased with 3 inches of gunite.

The December 1989 URS/Blume report describes the analyses of record for the stack building. The staff determined that the analyses satisfied the restart criteria. On the basis of these analyses, the following modifications were found to be required for the stack building:

- (1) Three 5-foot-thick shear walls must be added between four of the columns at the east end of the building. These shear walls span from the basement up to the crane girders.
- (2) The crane girders must be strengthened so that the end reinforcement of the girders provides continuity across the supports.
- (3) A tension tie must be added between the roof of the fan room and the main stack building.
- (4) A 6-foot-long saw cut must be made between the fan room column and wall.
- (5) The construction joint in the 5-foot-thick wall, located between two of the main columns near the fan room, must be strengthened.

Documentation of the implementation of the modifications for the K-area stack building have been received by DOE. This documentation, plus visual observation of the modifications, adequately resolves the issue concerning the 0.2g upgrade of the stack building.

URS/Blume determined that the bearing pressures under the stack building foundation were acceptable by performing a foundation capacity study. The DOE staff reviewed this work and determined that it satisfies the restart criteria.

Cooling Water Reservoir

The initial seismic analysis of the CWR structure was performed by Quadrex Corporation. All demand/capacity (D/C) ratios were equal to or less than unity. The critical element was found to be the lower end of the wall between the pump well and the pump house, where the D/C ratio was found to be equal to unity for the combined bending and axial load case as verified by DOE staff in an independent analysis.

The D/C ratio of unity warranted further analysis by WSRC. Based on this analysis (WSRC-RP-90-1338), which is described in Section 6.1.6, D/C ratios for moments and shears at all critical sections were evaluated to be less than unity. The maximum D/C ratio at the bottom of an exterior basin wall was 0.37 in flexure and 0.35 in shear. The maximum D/C ratio occurred in a pilaster in the center wall of the pumpwell of the CWR. This maximum value was 0.95 for flexure.

Effluent Sump Structure

Quadrex Corporation also performed the initial seismic analysis of the PES structure, which was discussed in Section 6.1.6 of this SER. The stress evaluation was conducted in the same manner as the evaluation of the cooling water reservoir. All of the safety margins were found to be equal to or greater than unity. The critical element is an exterior wall, where a safety margin of unity was found for the shear capacity.

This high D/C ratio evaluated by Quadrex dictated that WSRC perform additional analysis. This analysis, described in Section 6.1.6, indicated that all D/C ratios for flexure and shear at critical sections were less than one. The maximum D/C ratio was 0.94 for flexure at mid-span of an exterior wall.

Open Items

The following open item requires resolution prior to restart:

- o The floor slabs and foundation basemat in the reactor building must be evaluated for the seismic load case.

Concerning the remaining open issue listed above, WSRC has completed an analysis of the K-Reactor building floor frequencies and basemat stresses (WSRC-RP-90-1328). The DOE staff is in the process of reviewing this work.

6.1.8 Piping

This subsection evaluates the following piping: (1) Process Water System (PWS), including the Upflow Cooling Water System (UFCWS); (2) Supplementary Safety System (SSS); (3) Cooling Water Supply System (CWS); (4) Moderator

Recovery System (MRS); and (5) Contaminated Water Removal and Disposal System (CWRDS). The systems have been qualified by using dynamic analysis, equivalent static analysis, or the interim verification program procedure (WSRC's SEP-24, "Verification of Seismic Adequacy of Piping Systems," Rev. 2, December 12, 1990) using a span guideline and supplemented by static system analysis or response spectrum analysis, as appropriate. Experience-based walkdowns using screening criteria were also conducted for most of the piping systems and were used in the qualification of anchorage and small bore piping. Finally, soil deformation and wave passage analyses were performed for buried steel piping.

Process Water System Piping

The PWS consists of six primary coolant loops that circulate moderator to cool the reactor. Each loop consists of a main coolant pump, two parallel heat exchangers, expansion joints, and valves. All branch lines connected to the primary water system up to the first normally closed isolation valve require seismic qualification to assure pressure boundary integrity. This qualification has been demonstrated by various dynamic analyses or by the previously mentioned interim verification program procedure.

Portions of the PWS piping were analyzed by URS/JAB (Blume) and Quadrex. Blume and Quadrex performed seismic analysis of selected segments of PWS piping loops from the C-, K-, and P-Reactors. Blume used the 1980 ASME Section III, Subsection NB, code, and Quadrex used the 1974 Section III, Subsection NC code to perform the respective analyses. Quadrex also made a comparison of results with Blume by converting the NC analysis to NB and found it to be comparable. Upset stress limits were used in both the Blume and Quadrex analyses. The results of these analyses have been extended to all loops for all reactors by similarity arguments. The staff has evaluated these analyses and, where necessary, has performed additional analyses and sensitivity studies to show acceptability. In addition, the staff conducted a comparison of the codes used by Blume and Quadrex with the SER restart criteria and found that the reported results remain valid. The section of pipe that runs from the reactor tank to the pump was analyzed by Blume and Quadrex. This analysis demonstrates that this portion of the PWS line is seismically qualified.

Due to the flexibility of the heat exchanger and its supports, as identified in an analysis performed by the DOE staff, the previous analysis for the pump to heat exchanger and heat exchanger to reactor tank analysis did not meet the restart criteria. The DOE staff performed an additional analysis to demonstrate seismic qualification, i.e. a dynamic response spectrum analysis for systems 2 and 4 of the PWS, using the SER restart criteria. This analysis dynamically coupled the pump to heat exchanger line and the heat exchanger to tank line with the flexible heat exchangers. By similarity, this analysis can be extrapolated to the remaining portions of the K-Reactor PWS lines.

PWS lines other than the primary loops are currently being analyzed in accordance with the interim seismic verification program procedures (WSRC SEP-240 using a span guideline and supplemented by static system analysis or response spectrum analysis, when necessary. The codes used in the SEP-24

procedure are the 1986 B31.1 code equations and the stress intensification factors of the pre-1981 B31.1 code. The DOE staff has compared the 1981 B31.1 code with the 1974 ASME Section III code and found them to be comparable. However the use of stress intensification factor of a pre-1981 B31.1 code with the 1986 B31.1 allowables needs justification which is an open issue. Two of approximately 11 calculation packages were reviewed by the DOE staff and found to be acceptable. In addition, the DOE staff has reviewed this interim seismic verification program and found it to be generally acceptable with a few exceptions that have been judged not to affect the results of the analysis significantly. The DOE staff has provided comments on the implementation of SEP-24, and WSRC is working to resolve these open issues.

Although the combination of analyses for the PWS piping generally meet the SER restart criteria, several issues require resolution before K-Reactor restart.

Open Items

o SEP-24 Issues

The justification for the code allowables for SEP-24 must be provided. In addition the comments made by the DOE staff regarding the implementation of SEP-24 must be addressed.

o J-Bolt Anchorage

The use of grouted "j-bolts" is a common construction detail found in the SRS reactors. This anchorage is found on the three-way supports and on the elbow trunnion supports on the PWS lines. These bolts were originally installed by excavating the concrete and welding the bolts to the steel reinforcement. The bolts were then grouted in place, filling the excavated portion of concrete. The strength of this anchorage is not known and WSRC has initiated a testing program to verify the as-built condition and its adequacy.

o Guide Supports

Based on the confirmatory analyses, some of the vertical guide supports at the minus 20-foot elevation are overstressed. Furthermore, existing gaps in several guides make them inactive as seismic supports. Analysis has shown that the functionability of the guide is not required if the elbow trunnion support retains its integrity and the higher pump nozzle loads are acceptable.

o Elbow Trunnion Supports

The evaluation of the elbow trunnion supports demonstrates that the machine bolts for the connection detail must be at least 1 inch in diameter. As-built walkdowns have shown that the placement and size of these bolts do not conform to the design. These supports should be modified to reflect the design. WSRC has initiated the process to insure that the appropriate modifications are made.

Supplementary Safety System Piping

The SSS is a standby safety system that can be actuated automatically or manually to shut down the reactor from full power, without the safety rods or control rods, through the injection of a solution of gadolinium nitrate (a neutronic poison) into the reactor tank. The SSS consists of two essentially identical, redundant systems: the Near Side and the Far Side.

A dynamic analysis of record was performed by United Engineers & Constructors (UE&C) for the SSS in the K-Reactor building. The UE&C report, which is preliminary, documents the seismic analysis of the Far Side SSS piping for the K-Reactor. The 1983 B31.1 code was used in this analysis supplemented by the 1983 ASME Section III, Subsection NC for flanges. The faulted limits were used. UE&C concluded, as a result of the analysis, that the Far Side SSS for the K-Reactor meets the specified stress criteria provided that recommended support modifications are made to the supporting system. By similarity, UE&C concluded that the Near Side SSS for the K-Reactor also meets the specified stress criteria provided that similar modifications are made in accordance with UE&C recommendations.

The UE&C piping stress analysis model is based on walkdowns performed in mid-1989 and includes as-built piping configuration, support location, equipment location, and weights of all in-line components such as valves and flanges. The UE&C analysis used the computer program ADLPIPE-E to demonstrate compliance with the ASME/ANSI B31.1-1983 Power Piping Code. It used as input the envelope of the Housner response spectrum with 1 percent damping and the Regulatory Guide 1.60 response spectrum with 2 percent damping in accordance with Regulatory Guide 1.61 requirements, as-built dimensions, and actual support stiffness for nonrigid supports. The combination of deadweight, normal operating pressure (300 psi), and seismic loads was considered. Thermal analysis was not performed, since the system operates essentially at ambient temperature. The DOE staff compared the codes used in this analysis with the SER restart criteria and found that the codes are comparable. This analysis meets the SER restart criteria.

Open Items

The staff reviewed one UE&C report and concludes that the UE&C analysis demonstrates the acceptability of the K-Reactor SSS (both Far Side and Near Side) for reactor restart provided that the following items are completed:

- o The UE&C preliminary report is finalized and issued to DOE for review.
- o All of the UE&C recommendations, including the support modifications and the drawing revisions, for the Near and Far Side SSS are satisfied.
- o The system evaluations are revised to include a consideration of the thrust loads associated with system actuation. The thrust loads occur when the seismic triggers automatically actuate the injection system on a design basis earthquake (DBE).

- o All observations requiring action and categorized as startup items in the SEWS for the K-Reactor SSS system are satisfactorily resolved.
- o The design modification to the SSS that will allow the neutronic poison to be injected into the suction side of the primary pumps is analyzed and installed according to the restart criteria.

UE&C has reanalyzed both the Near and Far Side piping considering thrust loads. A final report on the analysis has been issued. The staff is in the process of reviewing the report.

Cooling Water System

The CWS provides for heat removal from the PWS. River water stored in the 25-million gallon 186 Basin is pumped into the reactor building through two large headers. These headers supply cooling water to the 12 heat exchangers. The CWS must be capable of providing decay heat removal for 72 hours.

The CWS is being analyzed in accordance with the interim seismic verification procedures described by WSRC SEP-24. These procedures use a combination of span guidelines, static system analyses, and response spectrum analysis. The codes used are the same as those used in the PWS analysis conducted under the WSRC SEP-24 procedures. Therefore, the same open issue discussed in the PWS section for codes also applies to the CWS. Three of approximately 20 calculation packages were reviewed by the DOE staff. Four similar supports were found to be overstressed when friction loads due to seismic loading are considered.

The DOE staff also performed dynamic response spectrum analyses for two pipe lines considered representative of the cooling water lines in the K-Reactor. The lines analyzed were the cooling water supply line for heat exchanger 4B and the Raw Water Inlet line for the Caterpillar Diesel Generator System. These evaluations were performed in accordance with the Class 2 equation of the ASME Section III 1974 version of the code using the faulted operating condition allowable, the same as the SER restart criteria.

The cooling water supply line is a carbon steel line extending 36 feet from the heat exchanger to the cooling water header. It has three substantial supports designed to carry the deadweight loads while accommodating pipe line motions. It is considered typical of the larger diameter close-coupled systems at SRS. For the cooling water supply line, the dynamic analysis modeled the effects of the heat exchanger flexibility. The heat exchanger supports are flexible and contribute to the seismic response. This in effect dynamically couples the CWS inlet lines with the flexible heat exchanger supports and the PWS piping. When this is considered, the most highly stressed location exhibited a stress of 63 ksi versus the faulted operating condition limit 28.8 ksi.

The raw water line is an 8-inch-diameter carbon steel line that starts at a 12-inch-diameter header, passes through wall penetrations and finally divides into three 6-inch-diameter branch lines. The line has deadweight supports at many locations with rod type hangers of various configurations. It is considered typical of the many small diameter lines at SRS that extend long distances and appear very flexible. The maximum stress for the raw water line was 26 ksi as compared to an allowable stress of 36 ksi occurring at the elbow adjacent to the 12-inch-diameter header. The pipe stresses for this line are within the Faulted Condition of Code allowable when the simultaneous action of pressure, deadweight, and seismic loads is considered.

The buried steel piping analyses conducted for the L-Reactor are being extrapolated to the K-Reactor based on similarity. However, the largest cooling water piping for the K-Reactor is larger than that of the L-Reactor. Also, the span length of the short leg, which connects to Building 190, needs to be determined for the K-Reactor in order to confirm similarity of the piping between the two reactors. Therefore, the extrapolation of the L-Reactor results to the K-Reactor may not be valid. This requires additional assessment.

Open Items

The following remain open items.

- o The codes for the SEP-24 piping analysis procedures needs to be justified and the comments by the DOE staff have to be addressed.
- o The overstress issue for the cooling water supply line due to the effect of heat exchanger flexibility must be resolved.
- o The validity of extrapolation for the buried steel piping must be determined.
- o The adequacy of the four supports due to friction loads must be addressed.

Moderator Recovery System

The MRS returns moderator that has leaked through the PWS pressure boundary from a sump to the reactor tank. It consists of two sump pumps, one moderator process tank, and piping that is primarily 6 to 12 inches in diameter. The MRS is designed to provide for intermediate-sized leaks on the order of 5 to 1000 gallons per minute.

Quadrex Energy Services performed an analysis of the MRS in the K-Reactor building. The piping was analyzed in three sections: from the sump pumps to the process tank, from the process tank to the reactor tank, and the vent line. The pumps and process tank were also analyzed separately. In the case of the process tank, this involved modeling the tank and its supporting structure. Quadrex used the ASME 1983 Section III, Subsection NC upset limits with the 1984 summer addenda. Subsection NF service level B was used for the supports. Also, 1980 B31.1 was used for part of the material data. The DOE

staff compared the codes used with the SER restart criteria and found them to be acceptable.

The DOE staff reviewed the Quadrex analysis report and determined that the analysis met the SER restart criteria with the following exceptions:

- o The MRS piping is connected to a pump and tank that do not meet the 33 Hz frequency requirement. The pump has a frequency of 23 Hz; the tank has a frequency of 14 Hz. Determination of the impact of these flexibilities is required to demonstrate that Code allowables for piping, supports, and equipment are met.
- o Supports K-20, K-21, K-26, and K-27 are subjected to moment loading, which was not considered. Moments must be estimated and the supports evaluated for these moments.

Analyses have subsequently been performed to resolve these issues. The DOE staff has reviewed the analyses and found them to be acceptable.

Evaluation of the Contaminated Water Removal and Disposal System

The CWRDS consists of two sump pumps, drain piping, vent piping, a storage tank and a retention basin surrounding the storage tank. Portions of the vent and drain piping are buried.

Process and cooling water system leakage from valves, pumps, and flanges is possible following a DBE. Therefore, seismically qualified water removal capability is required to prevent flooding of the DC PWS pump motors. A WSRC study determined the maximum leakage to be 50 gpm.

Open Items

The DOE staff reviewed the Blume analysis reports of the various components of this system and determined that the analyses meet the SER restart criteria, with the following exceptions:

- o The buried piping should have considered the effect of the compression, shear and Rayleigh waves acting simultaneously. The staff is in the process of determining if this issue has significant impact on the qualification.
- o While all piping in this system has been qualified by either response spectra or equivalent static computer analysis, supports have been qualified only by the seismic walkdown approach. Concerns regarding the walkdown approach are described in the Evaluation of Equipment Qualified by Walkdown contained in Section 6.1.9.

Evaluation of the Upflow Cooling Water Piping System

The UFCW provides cooling water from the PWS heat exchangers to the bottom of the reactor tank. There are two separate systems, one from the northside heat

exchangers and one from the southside heat exchangers. Both systems must be analyzed because they are not perfect mirror images of each other. Quadrex Energy Services performed an analysis of the UFCW in the K-Reactor building using the 1986 Section III Subsection NC faulted limits. The DOE staff compared the code used with the SER restart criteria and found it acceptable.

The DOE staff has reviewed the Quadrex analysis report and concluded that the analysis met the SER restart criteria with the following exceptions:

- o The UFCW piping is connected to the heat exchanger, which does not meet the 33 Hz frequency requirement to qualify as being rigid. Determination of the impact of the heat exchanger flexibility is required to demonstrate that code allowables for piping, supports, and equipment are met.
- o The Quadrex analysis shows high thermal loads on the circular manifold supports. Confirmation is required that loads on these supports were determined considering support flexibility and that the supports were qualified for these loads.

Analyses have subsequently been performed to resolve these issues. The DOE staff has reviewed the analyses and found them to be acceptable.

Diesel Power System

The DPS supplies power to drive the DC pump motors on the PWS to provide recirculation of moderator for core cooling. This system consists of caterpillar diesels, generators, and the electrical distribution system.

The DPS lines are currently being analyzed in accordance with the interim seismic verification program procedure, WSRC SEP-24. Two of approximately 12 calculation packages were reviewed by the DOE staff and found to be acceptable. The open issue regarding the use of codes and standards for the PWS also applies to the DPS.

6.1.9 Mechanical and Electrical Equipment

This subsection concerns the evaluation of the following mechanical and electrical equipment (including the electrical components associated with pumps and valves): (1) the PWS, including the Septifoil system; (2) the SSS; (3) the CWS; (4) the MRS; (5) the CWRDS; (6) the diesel generator system; (7) equipment for automatic actuation of the SSS; (8) filter compartment latches and chocks; and (9) instrumentation required for monitoring tank level, pump flow indicators, and bulk moderator temperature following a DBE.

Evaluation of the Material of Record

All equipment (except the reactor tank, Bingham pumps, heat exchangers, and recirculation pump diesel generator system) has been qualified by Seismic Qualification Utility Group (SQUG) methodology. The recirculation pump diesel generator distribution system includes components qualified by analysis and shake-table testing. The evaluation of the material of record identified

various inaccuracies and omissions. The majority of the findings relate to the fact that limited seismic analysis had been performed. For equipment with an existing seismic analysis, input loads (e.g., equipment nozzle loads) are unconservative and have been invalidated. Revised piping analysis results will form the basis for the nozzle loads used in equipment qualification.

Reactor Tank Assembly

The K-Reactor tank is a 15-foot-high cylinder with a diameter of about 16 feet. Six tapered inlet nozzles are uniformly spaced around the circumference of the coolant inlet plenum and six outlet nozzles are uniformly spaced around the circumference of the tank bottom. The plenum and top tube sheet are supported independently of the reactor shell (1/2-inch thick) and bottom tube sheet by a bearing ring which is supported by 12 beams anchored in concrete. The top tube sheet is supported above the main tank shell by a 3/16-inch-thick expansion ring. The tank shell and bottom tube sheet are supported by 12 bearing pad assemblies. The reactor contains positions for 597 fuel and target assemblies, which are supported by the water plenum and top shield.

The K-Reactor tank has not been specifically analyzed in previous work. Geometrically the P-, L- and K-Reactor tanks are virtually identical. From a pressure integrity standpoint, the tanks can be considered identical with a few minor differences, but these geometrical differences are judged not to be significant enough to invalidate the similarity assumption between the L- and K-Reactor tanks. However, similarities do not exist in tank weight due to differences in the reactor charges.

Quadrex analyzed the reactor tank assembly for the L-Reactor using the ASME 1983 Section VIII, Division 2 rules. However, the time history soil structure interaction input and analysis methods have not been reviewed against the restart criteria of this SER. Also, the DOE staff invalidated the nozzle loads used by Quadrex (except for the tank outlet nozzle) due to the issues of heat exchanger and pipe support flexibility.

The DOE staff conducted a confirmatory analysis in accordance with the SER restart criteria using a simple beam model for the reactor tank. It was found that the increased nozzle loads and the change in the floor response spectra input did not impact the conclusions derived from the Quadrex analysis. However, the weight assumed in the Quadrex analysis of the L-Reactor tank was not the same as that of the K-Reactor due to the reactor charge. The DOE staff conducted a confirmatory analysis on the K-Reactor tank assembly using the SER restart criteria.

No seismic stress analyses are reported for the reactor plenum inlet nozzle, upper plenum, top tube sheet, or reactor internals. The analysis for the top tube sheet support does not meet the SER restart criteria. The DOE staff conducted confirmatory analyses using the SER restart criteria to assess the seismic design adequacy of the inlet nozzle and the top tube sheet supports.

Based on its evaluation, the DOE staff determined that seismic input, analytical modeling procedures, and response determination of the confirmatory analyses meet the restart criteria. Except for the plenum inlet nozzle and

the reactor-internals issues discussed below, all stress levels meet the restart criterion.

Open Items

Although all seismic stress levels in the reactor components are believed to be within allowable values, the open items discussed below require resolution prior to restart.

- o No material of record demonstrating the seismic adequacy of the reactor internals was identified. The DOE staff has since performed a detailed preliminary analysis of the fuel and target assemblies. Although preliminary results from this analysis are satisfactory, this item is open pending documentation of the final results of this analysis.
- o Initial seismic analyses of the top tube sheet support and bottom tank support were inadequate. Preliminary results from the DOE staff confirmatory analyses demonstrate the adequacy of the supports. However, a significant input parameter for these analyses is the load from the fuel and target assemblies analyzed by the DOE staff. An analysis of top and bottom supports was performed that included loads from the fuel and target assemblies. Results of this confirmatory analysis still require documentation to close this issue.
- o The existing Quadrex analysis of the tank expansion ring was not validated for the K-Reactor tank. The DOE staff confirmatory analysis of the K-tank expansion ring shows its adequacy. This issue will be closed pending WSRC documentation of the results.

Heat Exchanger

- o The PWS heat exchanger transfers heat generated in the reactor tank to the CWS. The heat exchanger has a 0.50-inch, wall-thickness shell with an inside diameter of 87.5 inches and an approximate length of 28 feet. Twelve evenly spaced baffle sheets and two tube sheets support the heat exchanger tubes which are cored with solid rods to provide a narrow (50 mil) annulus for coolant flow. The heat exchangers are mounted on saddles, which are supported on wheeled carriages. The saddles are provided with seismic supports.

There is no seismic analysis of record for the heat exchanger nozzles, shell, and internals. Consequently, the DOE staff conducted confirmatory analyses on these items using the SER restart criteria. These analyses used nozzle loads derived from a coupled heat exchanger and a PWS piping confirmatory analysis, which accounted for flexibilities of the heat exchanger and the piping supports. The analysis of the inlet nozzle indicated a maximum stress of 14.5 ksi, which is 44 percent of the allowable. The fundamental frequency of the heat exchanger internals (tubes and baffle plates) was calculated to be greater than 33 cps. Thus, a rigid assumption can be used, which results in seismic stresses being less than one-third of the deadweight stresses. The tank shell is rigid by virtue of the baffle plates and the shell dimensions.

Based on its evaluation, the DOE staff concluded that seismic input, analytical modeling procedures, and response determination of the heat exchanger confirmatory analyses meet the SER restart criteria.

The SRS walkdown evaluation screening and evaluation work sheet (SEWS) assessed the heat exchanger supports based on the equivalent static load method. Due to the support flexibility issue, this evaluation was found to be inadequate. The DOE staff has performed a confirmatory analysis on the heat exchanger supports to account for the higher loads resulting from the heat exchanger support flexibility. The issue of heat exchanger support qualification remains open.

Open Items

Based on the above discussion, the following open item must be resolved prior to restart.

- o The DOE staff performed analyses to assess the supports and found that the stresses in several support components exceed the SER restart criteria. The operating contractor is currently addressing this open issue.

Bingham Pumps

Each of the six circulating loops of the PWS contains a Bingham pump. Although each pump is identical, nozzle loads differ due to differing piping configurations. Based on the 1983 URS/Blume analysis, similarities in the suction line piping were identified. It was determined that the six systems could be represented by only two sets of nozzle loads. An assessment confirming the similarities of the loops still needs to be checked.

A review of SRS qualification documents and discussions with SRS personnel determined that a SEWS package is the only material of record that exists for the qualification of the Bingham pumps. The SEWS concluded that the Bingham pumps are seismically adequate provided the identified minor construction modifications are implemented. The SEWS identified that a hoist chain could swing during an earthquake and possibly damage the lube oil lines. This deficiency along with the other changes identified in the SEWS, must be corrected prior to restart of the K-Reactor. A Design Request Modification (DRM) has been issued.

To confirm the material of record for the pumps and to provide additional assurance of pump seismic adequacy, the DOE staff performed an analytical evaluation of the K-Reactor PWS Bingham pumps using the SER restart criteria. This evaluation consisted of a dynamic finite element model that used nozzle loads from the 1983 URS/Blume analysis. Seismic loads used correspond to the set of higher nozzle loads. Results from the pump model indicated that the pump is dynamically rigid and resonance will not occur before, during, or after a seismic event. Since the pump is dynamically rigid, the structural response was evaluated using the equivalent static load method. Based on this evaluation, the Bingham pumps are seismically adequate and will perform their intended function during and after a seismic event.

As discussed in the PWS piping section, a new PWS piping analysis of the pump-to-heat-exchanger line has been performed by the DOE staff. The new pump outlet nozzle loads from this analysis will induce higher loads on the pump anchorage. Also, resolution of the heat exchanger supports overstress issue may further increase pump outlet nozzle and anchorage loads.

Open Items

Based on the above discussion of the seismic evaluation of the Bingham pumps, the open items discussed below require resolution prior to restart.

- o The assessment confirming the similarities in the suction line piping configurations for all six pumps must be checked.
- o The seven required construction modifications identified in the Bingham pump SEWS, "Modified NCR Forms," must be completed.
- o Thermal loads on the pump support were not assessed. An analysis of the thermal loads and the revised nozzle loads has been completed but has not yet been checked.

Active Valves

Active valves, which are required to operate during or after a DBE, must be assessed for operability. The DOE staff conducted a preliminary evaluation of the SQUG walkdown calculations conducted for the two valve types (PWS rotovalve and CWS motor-operated valve) that were affected by the piping confirmatory analyses results. The staff found that, although the valves are within the data base, the PWS rotovalve did not fit within the data base parameters for rigid valves. A confirmatory analysis demonstrated that the PWS rotovalve is rigid. However, the issue of valve operability remains open pending completion of the staff's review on all active valves.

The complete assessment of the SEWS for valves will be reported at a later date.

Recirculation Pump Emergency Diesel Generator System

The Recirculation Pump Building Emergency Diesel Generator System supplies emergency power for the sump pumps, effluent pumps, and moderator make-up pump and valves following loss of normal offsite power. Two generator units are installed for the K-Reactor site; each unit, rated at 800KW, is housed in its own metal enclosure adjacent to, but separate from the 107 Building. In addition, each generator unit has its own dedicated control system, transformer, diesel fuel oil (DFO) supply tank, and other auxiliary equipment.

The various portions of the diesel generator units and their auxiliary equipment were seismically qualified through analytical, testing, and experience data methods for the restart effort.

The equipment items in this system qualified by analysis include:

- o generator breaker support frame
- o diesel generator skid base anchorage
- o metal enclosure
- o DFO tank
- o exhaust silencer supports
- o anchorages
- o foundations

The diesel generators, themselves, and the control panel/cabinets were qualified through an experience data approach. For the analysis of the actual equipment items, the qualification analyses all used an equivalent static approach to develop seismic loads. These loads were then applied to calculate equipment and support critical stresses, anchorage loads, and foundation loads (where applicable). The codes used in the analysis include ASME section III, Subsection NC, AISC 8th edition and 1976 ACI-341.

Open Items

The DOE staff reviewed the qualification analyses for the various equipment items against the restart criteria. All items were found to be acceptable with the exception of the open items identified below. These items must be addressed prior to restart.

- o The qualification for the DFO tank used a Bijlaard (WRC 107) approach to analyze the shell stresses in the saddle region. However, the location of the saddles relative to the ends of the tank results in analysis parameters that are beyond the range of valid Bijlaard parameters. In addition, the analysis of one shell area resulted in a slight overstress that was judged acceptable. This judgement may be affected by the Bijlaard method used for this tank configuration. Additional justification is required to demonstrate tank adequacy.
- o The exhaust silencer is mounted atop the roof of the metal enclosure structure on the roof beams. A review or analysis of the roof beam connection to the enclosure structure was not done and is necessary to demonstrate the adequacy of the exhaust silencer mounting and the enclosure/beam connection.

The Recirculation Pump Emergency Diesel Generator System included several auxiliary components (not the diesel generator units themselves) which were seismically qualified by testing for the restart program. Examples of these equipment items are relays, the automatic transfer switch, disconnect switches, and transformers.

Documentation describing the test procedures and results were reviewed against the restart criteria by the DOE staff. The review included examination of the test and field mounting configurations, test versus required seismic inputs, and review of functionality requirements and/or monitoring, as required. All recirculation pump diesel generator units qualified by testing were found to satisfy the SER restart criteria.

Evaluation of Equipment Qualified by Walkdown

A SQUG-type walkdown uses procedures that incorporate historical records of equipment and piping performance in past earthquakes and accumulated test data as a baseline to verify the seismic adequacy of mechanical and electrical equipment in older operating nuclear power plants. These equipment records are incorporated into a generic implementation procedure (GIP) along with guidelines defining required training for seismic review team members, which equipment should be evaluated, equipment evaluation worksheets, anchorage requirements, and relay review guidelines.

Open Items

WSRC report RTR-2582 describes the SRS walkdown procedure. RTR-2582 contains equipment experience-data-based walkdown procedures that differ from the SQUG-GIP in some important respects. Because of these differences, the following open items must be resolved prior to restart:

- o There is no procedure for support member qualification.
- o The methods for addressing outliers are not defined for all equipment types.
- o There is no comprehensive document describing and controlling changes in the SQUG-type program.
- o Justification is not provided for the safety factors, but assessment is being made by WSRC to show acceptability.
- o Valve anchorage requirements are not provided.
- o Relay functionality using seismic experience data base is not acceptably demonstrated.

RESTART CRITERIA FOR THE SEISMIC EVALUATION

Geology

The restart criteria outlined below are a summary of the criteria found in sections 2.5.1 and 2.5.3 of the NRC Standard Review Plan.

WSRC shall describe the following:

1. Stratigraphy and geology of both the region and the site.
2. Structural geology of the site and evaluation of the engineering significance of geologic features underlying the site.
3. Capable faults in the region.
4. Evaluation of faults within the reactor site area including an evaluation of age of last movements.

Seismic Ground Motion

The restart criteria outlined below are a summary of the criteria found in Section 2.5.2 of the NRC Standard Review Plan and based on seismic criteria used in the NRC Systematic Evaluation Program.

WSRC shall meet the following criteria for vibratory ground motion.

1. Describe the seismicity at the site, including a listing of all earthquakes of magnitude 3 and greater within 200 miles of the site.
2. Describe the geologic and tectonic characteristics of the site, including a discussion of capable faults located within 200 miles of the site.
3. Correlate the seismic activity in the area with the geologic structures. (If such correlation is shown not to exist, seismic activity is to be correlated with the area tectonics.)
4. Determine the earthquake potential of geologic structures or tectonic provinces and describe the vibratory ground motion from maximum credible earthquakes.
5. Determine the seismic wave transmission characteristics of the site and define a free-field response spectral shape consistent with these characteristics.
6. Determine the Design Basis Earthquake (DBE) for the site.

7. Evaluate probabilistic seismic hazard results to determine adequacy of the DBE.

Geotechnical Engineering and Site Soil Conditions

The restart criteria outlined below are a summary of the criteria found in Section 2.5.4 of the NRC Standard Review Plan.

WSRC shall meet the following criteria for soil properties and behavior.

1. Describe the in-situ properties (static and dynamic) of the soils supporting the facility.
2. Describe the engineering significance of the geologic history of the site as it effects the in-situ properties (static and dynamic) of the soils supporting the facility.
3. Describe the engineering significance of the site soil features as they relate to determining potential zones of weakness or alteration due to long-term weathering processes or due to man's activities in the area.
4. Describe the ground water conditions at the site and its potential impact on soil behavior and evaluate the potential for soil settlement and liquefaction under seismic loads associated with the DBE.
5. Describe the response of site soil to dynamic loading.

System Response to Seismic Motions

The restart criteria outlined below are generally a summary of the criteria found in Section 3.7 of the NRC Standard Review Plan.

WSRC shall meet the following criteria concerning seismic design parameters and analysis.

1. The free field time history, used in the Soils Structures Interaction (SSI) analyses, shall envelop the criteria free field spectra for all damping values either used in the SSI analyses or in the application of the floor response spectra (FRS). The enveloping criteria shall be in accordance with the acceptance criteria of Section 3.7.1 of the Nuclear Regulatory Commission's (NRC's) NUREG 0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP). This requires that the computed spectra cannot dip below the criteria spectra by more than 5 percent; that a visual inspection of the two spectra shows no wide frequency bands where the computed spectra is below the criteria, and the power spectral density (PSD) functions are examined to ensure sufficient energy at all frequencies.
2. The structural and equipment damping values used for the analyses shall be in accordance with NRC Regulatory Guide (RG) 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." Piping systems shall use RG 1.61 damping values unless dynamic analyses are performed; then, the

values may be taken from ASME Code Case N-411-1, "Alternative Damping Values for Seismic Analysis of Piping Section III, Division 1, Class 1, 2, and 3" (February 1986). Soil damping may be based on generic data for the soil types found at the site.

3. The SSI models shall properly include the following effects: frequency dependence, layering effects, depth of foundation embedment, depth to bedrock, soil property degradation as a function of soil strain, ground water level, and building to building interaction.
4. The SSI analyses shall be based on a range of static soil properties that are consistent with the available data at the site, and on the free field criteria motion applied at the surface. Demonstrate that the attenuation of this motion over the depth of the structures is not excessive and is consistent with the soil properties at the site.
5. The structural model shall be demonstrated to adequately reflect the vibrational characteristics of the structure. Vibrational mode shapes and frequencies shall be provided.
6. The time increment for a time domain analysis or the frequency increment for a frequency domain analysis shall be demonstrated to be sufficient to pick up the frequencies of interest.
7. Torsional effects shall be included in the analyses or demonstrated to be negligible.
8. The methods used to broaden the peaks and smooth the calculated FRS shall be sufficient to account for reasonable uncertainties in the model.

Structures and Foundations

WSRC shall meet the following criteria for the (1) reactor buildings, (2) stack building, (3) cooling water reservoir, and (4) process effluent sump

1. Demonstrate that all structures and foundation elements satisfy the ultimate load requirements of the American Concrete Institute (ACI) Code (ACI 349-85, "Code Requirements for Nuclear Safety-Related Concrete Structures", 1985) and the American Institute of Steel Construction (AISC) Code (Manual of Steel Construction, 7th Edition, June 1973), when the design basis event (DBE) (corresponding to the RG 1.60 spectra anchored at 0.2 Gs) seismic loads are combined with the operating loads using a load factor of unity for all load components.
2. Demonstrate that the contact pressure under foundation elements is less than the ultimate capacities of the soils at the site when the DBE seismic loads are combined with normal operating loads.

PIPING

Piping Qualified by Dynamic Analysis

WSRC shall meet the following criteria for piping that is qualified by dynamic analysis.

1. Input Definition

The criteria stated in the System Response to Seismic Motions section of the seismic criteria shall be used to develop the input floor response spectra (FRS) for analysis.

A. Horizontal Input

The 1 percent damping Housner free field response spectrum anchored at 0.2g and amplified by 10 percent shall be used as the FRS for piping systems that have a dynamic analyses of record. For any new dynamic analyses of piping systems, the FRS developed from R.G. 1.60 free-field response spectra shall be used. If other FRS are used, they shall be justified.

B. Vertical Input

As a minimum, two-thirds of the horizontal FRS shall be used as the vertical FRS unless justification is provided to show that another value produces negligible differences in the response of interest (e.g., displacements, loads, and stresses).

2. Procedures Used for Analytical Modeling

The stiffness, mass, and damping characteristics of the as-built system shall be adequately incorporated into the analytical models as described below.

A. As-Built Data

The analyses shall be based on as-built information obtained from as-built or SQUG walkdowns.

The design drawings, which are the bases for existing analyses reported in the material of record shall be checked against the as-built drawings. Design drawing data may be acceptable if the difference between the design drawings and the as-built drawings is within the tolerance limits discussed in PVRC WRC Bulletin 316, "Technical Position on Piping System Installation Tolerances" (July 1986).

B. Piping Support Stiffness

Rigid supports shall be modelled as rigid restraints. Supports are rigid if the analysis results are not significantly different

from results obtained using the actual stiffness. Supports with stiffness values comparable to the recommendations in Lin and Chu, Paper #49, PVP Vol. 155 (1989) can be considered as rigid. Alternatively, the actual support stiffness may be modelled and used in the piping model.

C. Equipment and Equipment Support Stiffness

Piping shall be coupled to equipment if the equipment or its support is not rigid (e.g., less than 33 cps) in any given direction, unless the effect of stiffness is shown to be negligible. Alternatively, the equipment nozzle displacement may be applied to the attached piping as piping seismic anchor movement (SAM).

D. Damping

Use 1 percent damping when Housner FRS is used. Use Reg. Guide 1.61 or ASME Code Case N-411 damping when Reg. Guide 1.60 based FRS is used.

E. Similarity Criteria

Similarity shall be assumed for two systems if the difference in dimensions is within the acceptable tolerance limits set forth in PVRC WRC Bulletin 316 , Technical Position on Piping System Installation Tolerances, (July 1986).

Two pieces of equipment are similar if the operating conditions, material properties, equipment configuration, and support dimensions and arrangements are the same.

Two pieces of equipment can be assumed to be similar even with minor dimensional differences provided that both pieces of equipment are either rigid or differ by less than ± 10 percent in the fundamental natural frequencies.

F. Decoupling Criteria

The run pipe may be decoupled from the branch pipe when the moment of inertia ratio of the two is larger than 25, per WRC Bulletin 300 Technical Position on Industry Practice, (December 1984).

G. Mass Modeling

An adequate number of masses or degrees-of-freedom shall be used in modeling to determine the dynamic response. The number of degrees-of-freedom shall be at least equal to twice the number of modes with frequencies less than 33 cps. Alternatively, each piping span must have enough masses between supports to predict frequencies of at least 33 cps.

3. Response Determination

A. Consideration of Modes

A sufficient number of modes shall be used to ensure participation of all significant modes, if the modal analysis approach is used. The number of modes is sufficient if at least one rigid frequency mode (at or above 33 cps) is included. Justify the integration time steps used if the analysis is based on time history direct integration method.

B. Combination of Modal Responses

The most probable total response shall be obtained as the square-root-sum-of-the-squares (SRSS) of the response from individual modes, when using the response spectrum method. Use either modal superposition or direct integration approach for time history analysis.

C. Three Components of Earthquake Motion

Use the SRSS approach to combine maximum codirectional responses caused by each of the three components of earthquake motion at a particular point in the mathematical model when the response spectrum method is used. Use linear superposition for time history analysis.

4. Code Allowables

A. Piping and Piping Components

The code of reference shall be based on ASME Boiler and Pressure Vessel Code Section III, 1974 edition including the Winter Addenda.

The allowable stress limits under faulted operating conditions specified in ASME Code Case 1606 (Stress Criteria, November 5, 1973) shall be used for piping.

The combination of elbow stresses and pipe stresses due to the lug shall be used to qualify elbows with welded attachments.

NC-3647 of ASME Code Section III shall be used for flanges.

Pipe stresses from support anchor displacements shall be computed using the requirements of NC-3652 of ASME Section III.

ASME Section III, Code Class 2 rules shall be used for expansion bellows.

B. Supports

The 1973 AISC Code (AISC Manual of Steel Construction, 7th Edition, June 1973) shall be used for support evaluation.

Use EPRI data (EPRI NP-5228, Seismic Verification of Nuclear Plant Equipment Anchorage, Volume 1, Development of Anchorage Guidelines, May 1987) converted to the 2500 psi concrete for cast-in-place and grouted-in-place bolts, expansion, lead shield and lead cinched anchors. Use safety factors of 3 and 4 for non-shell and shell expansion anchors, respectively. Use 1973 AISC Code (Manual of Steel Construction, 7th Edition) requirements for steel connecting members and bolts. Use ACI Standard 349-85 (Code Requirements for Nuclear Safety Related Concrete Structures, 1985) for concrete evaluation.

C. Expansion Joints

The manufacturer's allowable displacement limits for expansion joints for the existing expansion joints (see Hartman, et al, Evaluation of Expansion Joints in the Reactor Process Water Piping, DuPont, July 1988) and the new Model X expansion joints (see Specification for Procurement of Metallic Bellows-Type Expansion Joints in the Reactor Process Water, Westinghouse, July 1989) shall be used. Use NC-3649 as the acceptance criteria for stress computation if the manufacturer's allowable displacement limits cannot be met and code stress computation is used to show acceptability.

5. Special Requirements

A. Interaction of Other Systems with Category I Systems

Procedures for evaluating interaction are part of the SQUG Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Revision 0, June 1988.

B. Multiply-Supported Equipment and Components with Distinct Inputs

- (i) Either Independent Response Spectra, Envelope Response Spectrum or time history base motions shall be used to compute seismic stresses due to inertia loads.
- (ii) The maximum relative displacements among supports of Category I structures, systems, and components shall be computed by including the effect of the equipment and building if it is deemed to be significant (i.e., if it increases response by 10 percent or more).

Seismic Anchor Movement (SAM) is not to be included if the piping model includes the equipment or building.

- (iii) Piping stresses from SAM shall be computed using the requirements of NC-3652 of the ASME Section III Code.
- (iv) The combinational approach for stresses from inertia loads and SAM shall be based on the SRSS approach.
- (v) The allowable stress limits defined in the Code Allowables section of this criteria under Piping and Piping Components shall be used.

C. One-Way Supports

Justify the linear analysis used for one-way supports (e.g., thrust blocks) as two-way restraints, by showing that the maximum gap between the pipe and support obtained from the linear analysis is less than 1/8 inch such that the support can still be considered as a restraint. Alternatively, the piping segments with one-way supports may be qualified by bounding results from both active (two-way) and inactive (no restraint) assumptions.

Piping Qualified by Equivalent Static Load

WSRC shall meet the following acceptance criteria for piping qualified by the equivalent static load method.

A. Simplified Model

WSRC shall justify that the system can be realistically represented by a simple model when one is used.

B. Modeling Approach

Include the criteria from all paragraphs in sections on Input Definition; Procedures Used for Analytical Modeling (except Mass Modeling); Interaction of Other Systems with Category I Systems; and Multiply-supported Equipment and Components with Distinct Inputs.

C. Static Coefficient

A factor of 1.5 shall be applied to the peak acceleration of the FRS defined under Input Definition in Piping Qualified by Dynamic Analysis to obtain an equivalent static load of a piping span. The use of lower static coefficients or FRS shall be justified.

D. Response Determination and Evaluation

The restart criteria of all paragraphs described in the sections on Code Allowables and Special Requirements (above) shall be used for response determination and evaluation. The criteria from the section on Three Components of Earthquake Motion shall also be used.

Buried Steel Piping

WSRC shall meet the following acceptance criteria for buried steel piping.

A. Relative Movements of Anchor Points

The effect of maximum relative dynamic movement between anchor points and the adjacent soil shall be included when evaluating the relative movements of anchor points. The motion of the anchor points shall be considered as out-of-phase, so as to result in an absolute sum determination of the relative displacements.

B. Pipe Movements Away from Anchor Points

For straight pipes away from anchor points, sharp bends, or intersections, the pipe strain shall be calculated by either ignoring or considering the soil friction as shown in equations 3500-1 and 3500-2 of ASCE 4-86, 1986. It is acceptable to use the lower of the strain values calculated by these two equations.

C. Pipe Movements at Bends, Intersections and Anchor Points

For bends, intersections and anchor points, the force shall be calculated as the product of the lower value of the maximum strain calculated from Equations 3500-1 and 3500-2 of ASCE 4-86, 1986, the piping area, and the secant modulus of the pipe.

D. Response Determination

The bending moments and shear forces shall be determined in the buried pipe using the force determined in Support Qualification above and formulas based on the theory of a beam on an elastic foundation. (See Hetenyi, Beams on Elastic Foundation, University of Michigan Press, 1974).

E. Response Combination

The results of anchor point movements from Relative Movements of Anchor Points above shall be combined with the pipe response computed in either Pipe Movements Away from Anchor Points or Response Determination above by SRSS.

F. Pipe Allowables

Use the piping yield stress as the allowable stress.

MECHANICAL AND ELECTRICAL EQUIPMENT

Equipment Qualified by Dynamic Analysis

WSRC shall meet the following acceptance criteria for equipment qualified by dynamic analysis.

A. Input Definition

The criteria stated in the System Response to Seismic Motions section of this criteria shall be used to develop the input floor response spectra (FRS) for the analysis of mechanical and electrical equipment.

1. Horizontal Input

The 1 percent damping Housner free field response spectrum anchored at 0.2g and amplified by 10 percent shall be used as the FRS only for equipment on which a dynamic analysis of record has already been performed. Use the FRS developed for the restart program (RP) (URS/JAB 6193, 06, "Floor Acceleration Response Spectra for Updated DBE and Reg. Guide 1.60 Seismic Input, Building 105K, Savannah River Plant," August 1989) for new analyses conducted after February 1989. Justify other FRS used.

2. Vertical Input

A minimum of two-thirds of the horizontal floor response spectrum shall be used as the vertical floor response spectrum unless justification is provided to show that another value produces negligible differences in the response of interest (e.g., displacements, loads, and stresses).

B. Procedures Used for Analytical Modeling

The stiffness, mass, and damping characteristics of the as-built systems shall be adequately incorporated into the analytical models as described below.

1. As-Built Data

The analyses shall be based on as-built information obtained from as-built or SQUG walkdowns.

The design drawings, which are the basis of existing analyses reported in the material of record, shall be checked against the as-built drawings. Design drawing data will be acceptable if the difference between the design drawings and the as-built drawings is within the tolerance limits set forth in PVRC WRC Bulletin 316 (July 1986).

2. Equipment and Equipment Support Model

(a) Equipment Internals

Flexible internals shall be included in the equipment model unless it can be shown that the equipment

internals and the equipment (including support) are sufficiently decoupled such that they can be analyzed separately. They are sufficiently decoupled if the frequencies differ by at least a factor of 2.

(b) Equipment Body

The equipment shell mode (such as the reactor tank shell) shall be included in the model if it is deemed to be flexible (less than 33 cps). Consider only the non-symmetric shell modes.

(c) Equipment Support

Equipment support or equipment support stiffness shall be included in the analysis model if the equipment support is not rigid, e.g., less than 33 cps in any given direction when equipment mass is considered, unless the effect of stiffness is shown to be negligible.

(d) Connecting Piping

Piping may be decoupled from the equipment model.

Equipment nozzle loads obtained from the piping analysis shall be used in the equipment and equipment support analysis. The equipment nozzle loads shall be obtained from the coupled equipment and piping analysis if the equipment or equipment support is flexible.

The loads at the piping anchor to the equipment nozzle shall be used as the equipment nozzle loads if the equipment and its supports are rigid, or if the equipment nozzle stiffness is modeled in the piping analysis.

3. Similarity Criteria

Two pieces of equipment are similar if the operating conditions, material properties, equipment configuration, and support dimensions and arrangements are the same.

Two pieces of equipment can also be assumed similar with minor dimensional differences such that both equipment are either rigid or differ by less than ± 10 percent in the fundamental natural frequencies.

4. Mass Modeling

An adequate number of masses or degrees-of-freedom shall be used in modeling to determine the dynamic response. Ensure that the number of degrees-of-freedom is at least equal to twice the number of modes with frequencies less than 33 cps. Alternatively, the equipment must have enough masses to predict frequencies of at least 33 cps.

5. Damping

A damping value of 1 percent shall be used for equipment when analyses of record have been performed on the basis of Housner Response Spectrum with a 1.10 factor.

A damping value of 3 percent shall be used for new analyses for pressure retaining equipment including valves, pumps, tanks, and heat exchangers, etc., as discussed in RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," October 1973.

A damping value of 5 percent shall be used for new analyses for non-pressure retaining parts of equipment, such as motor control centers, cabinets, panels, racks, fans and transformers, etc. (See IEEE-344-75 and "Restart Criteria for Seismic Qualification", a memorandum from Patterson to Peckinpugh, dated November 1988). Higher damping values may be used for special designs when justified.

C. Response Determination

1. Consideration of Modes

A sufficient number of modes shall be used to ensure participation of all significant modes. The number of modes is sufficient if at least one rigid frequency mode (at or above 33 cps) is included.

Justify the integration time steps used if the analysis is based on time history direct integration method.

2. Combination of Modal Responses

The most probable total response shall be obtained from the square-root-sum of the squares (SRSS) of the individual modal responses, when using the Response Spectrum Method. Use either modal superposition of three components of earthquake motion or direct integration approach for time history analysis. The integration time step shall be justified if the time history direct integration method of analysis is used.

3. Three Components of Earthquake Motion

Use the SRSS approach to combine maximum codirectional responses caused by each of the three components of earthquake motion at a particular point in the mathematical model when the response spectrum method is used.

Use linear superposition for time history analysis.

D. Special Requirements

1. Interaction of Other Systems with Category I Systems

Procedures for evaluating interaction are part of the SQUG-GIP. GIP acceptance criteria are discussed in the section on Equipment Qualified by SQUG Walkdown Procedure, below.

2. Multiply-Supported Equipment and Components with Distinct Inputs

(a) Either Independent Response Spectra, Envelope Response Spectrum, or time history base motions shall be used to compute seismic stresses due to inertia loads.

(b) The maximum relative displacements shall be computed among supports of Category I structures, systems, and components by including the effect of the equipment and building if it is deemed to be significant (i.e., if it increases response by 10 percent or more).

Support relative displacements shall not be included if the equipment model includes the building.

(c) Equipment stresses from support anchor displacements shall be included using the requirements of NC-3000 of the ASME Section III Code.

(d) The combinational approach for stresses from inertia loads and support relative displacements shall be based on the PVRC Task Group report, "Independent Support Method of Modal Spectral Seismic Analysis" (draft, September 11, 1989).

(e) The allowable stress limits shall be as defined in the Code Allowables section discussed below.

3. Determination of Number of Earthquake Cycles

Earthquake cycles need not be considered in the seismic qualification for the reactor restart.

4. Other Loadings

(a) Water Sloshing

ASCE 4-86, 1986 shall be used to compute water sloshing loads and impulsive loads due to seismic motion for large tanks (such as reactor tank and the water storage tank). The effect of these loads and the hydrostatic pressure loads shall be added and then combined with the effect of the seismic inertia loads by absolute sum.

(b) Valve Operation Thrust Loads

The effect of the operational thrust loads due to valve actuation shall be calculated and combined with the effects of the thrust loads with other operational loads.

5. Equipment Operability

For active equipment (i.e., the equipment which is required to operate during and after a DBE), the analysis results shall demonstrate that the equipment operability is maintained. This includes demonstrating that

- (a) contact of the moving parts (such as rotor) with the stationary parts (such as stator) will not occur;
- (b) bearing loads are within the manufacturers' allowables. Typically, bearing loads are acceptable if the shaft seismic accelerations are less than 3g and 2g, for the horizontal and vertical directions, respectively;
- (c) valve accelerations are within the manufacturers' limits. Typically, valve accelerations shall not exceed 3g and 2g in the horizontal and vertically directions, respectively; and
- (d) nozzle loads are within the manufacturers' allowables. Typically, the nozzles of active equipment such as valves and pumps are much stronger than the pipe. Therefore, checking piping stresses at the piping anchor to the equipment nozzle alone is acceptable.

There are no operability requirements for the inactive equipment (i.e., equipment which is not required to function during and after a DBE).

E. Code Allowables

1. Pressure Retaining Parts

Use Faulted Operating Condition allowable stress limits from ASME Section III, 1974 Boiler and Pressure Vessel Code including the Winter Addenda for both active and inactive equipment.

(a) Shells, Internals, and Nozzles

The allowable stress limits defined in NC-3100 through 3500 of the Code shall be used.

Check potential for buckling whenever necessary.

(b) Flanges

The allowable stress limits defined in NC-3647 of the Code shall be used.

For new equipment analysis performed after June 1990, ASME Section VIII (ASME Rules for Construction of Pressure Vessels, Division 2, 1986), may be used in lieu of the ASME Section III. The allowable stress limits shall be based on $K=2$ for the faulted condition.

2. Non-Pressure Retaining Parts

For the non-pressure retaining parts including valve yokes, connecting bolts for valves and pumps, and equipment supports, use the allowable stress limits (including buckling requirements) in AISC Code 7th edition (1973) with an increase of $1/3$ (Section 1.5.6 of NUREG-0800, Rev. 1) for the DBE. Alternatively, the 8th Edition of the AISC Code, 1980 may also be used for analysis conducted after June 1990. Use ASME Section III Appendix F allowables for faulted condition loads when justified.

Use the Section on Anchorage for Equipment Qualified by SQUG Walkdown Procedure below for equipment anchorages.

Equipment Qualified by Equivalent Static Load Method

The acceptance criteria from the appropriate sections of the Equipment Qualified by Dynamic Analysis section shall apply to the equipment qualified by the equivalent static load method (a special case of the dynamic analysis method), as supplemented by the following:

A. Simplified Model

Justify that the equipment can be realistically represented by a simple model.

B. Static Coefficient

A factor of 1.5 shall be applied to the peak acceleration of the FRS defined in the Input Definition section of Equipment Qualified by Dynamic Analysis to obtain an equivalent static load if the equipment natural frequency is not calculated.

A factor of 1.5 shall be applied to the highest spectral value above the calculated equipment natural frequency. If a factor less than 1.5 is used it shall be justified.

If the FRS does not meet the criteria of the Equipment Qualified by Dynamic Analysis section, it shall be justified.

Equipment Qualified by SQUG Walkdown Procedure

The following restart criteria, generally taken from the SQUG-GIP and NUREG-1211, "Regulatory Analysis for Resolution of Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Plants," February 1987 shall be used to evaluate the SRS seismic restart program for equipment that has not been dynamically analyzed. These acceptance criteria identify the applicable equipment experience data, design caveats to avoid, and inputs and methods for qualifying outliers, supports, anchorages, and synergistic effects for each type of equipment included in the program.

A. Equipment Applicability

The equipment information contained in Table A shall be used to define the applicable equipment. (Table A is defined in the SQUG-GIP and modified by the November 1988 memorandum from Patterson to Peckinpah, "Restart Criteria for Seismic Qualification.")

B. Design Caveat

The NRC, SSRAP and SRS Walkdown Training Programs have specified special design conditions to avoid with respect to seismic qualification of equipment. SQUG experience data has likewise identified additional conditions to avoid.

These conditions shall be considered when qualifying equipment by walkdown.

C. Outliers

The methods used to show acceptability of outliers shall be identified. The acceptable methods include response spectrum analysis, equivalent static analysis, or testing.

D. Input Definition

1. Equipment Integrity

Use the free field response spectrum to compare with the bounding spectrum, if the equipment is at least 8 cps. Use the FRS to compare with the bounding spectrum times 1.5, if the equipment is less than 8 cps.

2. Equipment Functionality

Use the FRS developed for the RP to compare with the Generic Equipment Ruggedness Spectrum (GERS) specified in SQUG-GIP, as appropriate. For active equipment such as valves, the GERS defined is for the attachment point of the valve to the piping system.

3. Relay Functionality

Active relays shall be tested using a test response spectrum which envelops the FRS developed for RP and increased to account for the mounting structure on equipment and amplification, if any.

4. Equipment Anchorage

Use the equipment frequency and damping from testing or analysis to calculate the seismic demand for anchorage analysis. Use the data in Table A if the frequency is not known.

E. Support Qualification

The calculation method for equipment support stresses shall be based on the approaches recommended by the SQUG-GIP.

F. Anchorage

The evaluation approach for anchorages shall be based on the SQUG-GIP, and modified as follows.

Use EPRI data (EPRI NP-5228, Seismic Verification of Nuclear Plant Equipment Anchorage, Volume 1, Development of Anchorage Guidelines, May 1987) converted to 2500 psi concrete for cast-in-place and grouted-in-place bolts, expansion, lead shield and lead cinched anchors. Use safety factors of 3 and 4 for non-shell and shell expansion anchors, respectively. Use 1973 AISC Code (Manual of Steel Construction, 7th Edition) requirements for steel connecting

members and bolts. Use ACI Standard 349-85 (Code Requirements for Nuclear Safety Related Concrete Structures, 1985) for concrete evaluation.

G. Synergistic Effects

All synergistic effects shall be identified including the potential impact due to block walls, transit walls, masonry walls, loosely restrained heavy chains, non-Category I tanks, other equipment, and excessive piping motion. Calculation or engineering judgment shall be used to justify that the synergistic effects identified are acceptable.

H. Relay Functionality

A review shall be performed to identify equipment that may be adversely affected by relay malfunction. This shall include both active and passive electrically controlled or powered equipment which should not inadvertently start, stop, or change state due to relay chatter (or other malfunction of relays or other electrical contactor devices) in order to accomplish its safe shutdown function. Relay functionality shall be demonstrated by testing.

Qualification by Testing

A program shall be developed by WSRC which describes how the following acceptance criteria shall be used for the safety evaluation of newly purchased equipment and old equipment which is qualified by testing. This program shall require DOE approval.

- A. Seismic test (input) spectra or required response spectra (RRS) shall be more conservative than the FRS defined in the System Response to Seismic Motions section of this criteria.
- B. Either IEEE-344-75, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations" (1975) or IEEE-344-87, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations" (1988) shall be used for test requirements. Exceptions are provided below.
- C. Testing for OBE is not required.
- D. Testing for environmental loading conditions (e.g., temperature, humidity, radiation, material aging based on IEEE-323) shall be addressed as part of the long term environmental qualification program.

Table A
EQUIPMENT INFORMATION

<u>EQUIPMENT TYPE</u>	<u>TYPICAL WEIGHT</u>	<u>BASIS FOR DETERMINING SEISMIC DEMAND</u>
Motor Control Centers	625 lbs per cabinet	Peak of 5 percent damped demand spectrum
Battery Charger and Inverters	45 lbs pcf	Peak of 5 percent damped demand spectrum
Horizontal Pumps with Motors	<u>H.P.</u> 1,000 20,000 600 16,500 500 12,000 400 8,600 200 6,000 100 3,600	ZPA of demand spectrum
Vertical Pumps with Motors	<u>H.P.</u>	<u>Weight</u>
a. Vertical Immersion	150	4,000
		Peak of 2 percent damped demand spectrum
b. Centrifugal	500 9,000 2,000 48,000	
		ZPA of demand spectrum
c. Deep-Well	500 9,000 (motor) 14,000 (impeller)	
Transformers	<u>KVA</u> 3,000 15,000 2,500 11,050 2,000 9,400 1,000 6,300 100 975	ZPA of demand spectrum
Compressors with Motors	<u>H.P.</u> 50 4,000 200 10,000	ZPA of demand spectrum
Instrument Racks	20 pcf of vertical face	Peak of 5 percent damped demand spectrum

Table A continued

<u>EQUIPMENT TYPE</u>	<u>TYPICAL WEIGHT</u>	<u>BASIS FOR DETERMINING SEISMIC DEMAND</u>
Low-Voltage Switchgears or	35 pcf	Peak of 5 percent damped demand spectrum
Metal-Clad Switchgears	31 pcf	
Battery Racks	0.11 pounds per cubic inch for batteries, plus weight of racks	Peak of 5 percent damped demand spectrum
Generic Equipment Cabinets	3 times the weight of cabinet housing	Peak of 5 percent damped demand spectrum
Walk-Through Control Panels	Determine and use weight per foot of length	Peak of 5 percent damped demand spectrum

6.2 Core Issues

6.2.1 Core Design

6.2.1.1 Background

In the past, the responsibility for the core design at SRS was shared among contractor groups. WISR 23 recommended that the responsibility to understand and defend the core charge analyses resides in the Reactors Operations organization. The core design restart criteria require the Reactor Operations Division to have the ultimate responsibility for core design accountability. The restart criteria also require increased formality in the procedures and activities relating to core design.

6.2.1.2 Restart Criteria

The criteria for determining the adequacy of the SRS core design process for restart are described below.

(1) Responsibility

The responsibility for concluding that core safety evaluations are complete and reflect current plant conditions, as well as for defending these evaluations, shall be clearly defined as the responsibility of Reactor Operations Division. In addition, the necessary interface agreements with the laboratory support group shall be established.

(2) Design Requirements

- (a) Fuel design limits shall be determined that ensure that fuel integrity is maintained during anticipated and postulated reactivity and power transients, inside and outside the normal operating envelope.
- (b) The reactor core shall be designed with appropriate margins to ensure that acceptable fuel design limits are not exceeded during any condition of normal operation.
- (c) The reactor core shall be designed such that, in the power operating range, the net nuclear feedback characteristics tend to compensate for rapid insertions of positive reactivity.
- (d) The reactor core shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily suppressed.
- (e) The design of the core shall be such that either of two required reactivity control systems is capable of rendering and maintaining the core subcritical by at least the amount designated any time during active core duty.

- (f) WSRC shall demonstrate that all major reactivity contributions within the core are well understood and predictable in order to ensure confident reactor operation.

(3) Design Methods

- (a) The computer codes used in the core design shall be described in sufficient detail to establish that the theoretical bases, assumptions, and numerical approximations for a given code are established in accordance with Section 5.8 of WSRC-RP-89-1249 (Reference 1).
- (b) The analysis methods used in the core design shall be described in sufficient detail to establish that the theoretical bases, assumptions and approximations for a given analysis reflect or are comparable to current accepted industry standards and practices. (Reference requirements for computer code verification in Section 5.8).
- (c) The calculational methods used in the core design shall be described in sufficient detail to establish that the theoretical bases, assumptions, and approximations for a given calculation reflect or are comparable to current accepted industry standards and practices. (Reference requirements for computer code verification in Section 5.8).
- (d) The computer models used in the core design shall be described in sufficient detail to establish that the theoretical bases, assumptions, and approximations for a given model reflect or are comparable to current accepted industry standards and practices. (Reference requirements for computer codes verification in Section 5.8.)
- (e) The uncertainties associated with the core design analyses shall be clearly stated and evaluated and shall be acceptable with respect to current industry standards. (Software verification and validation shall be accomplished as discussed in WSRC-RP-89-1249, as discussed in Section 5.8.)

(4) Design Procedures

- (a) The administrative procedures used to generate the core design shall be described in sufficient detail to establish that all required interfaces and approvals reflect or are comparable to the current quality assurance standards.
- (b) The technical procedures used to generate the core design shall be described in sufficient detail to establish that all design inputs, methods, exceptions, and outputs reflect or are comparable to the current approved quality assurance standards.

- (c) The procedure for documenting the core design effort provides for sufficient detail to establish the credibility of the executions of the design methods and procedures.

(5) Design Products

- (a) The applicable core design parameters shall be clearly translated into fabrication requirements and transmitted to the proper departments.
- (b) The applicable core design parameters shall be clearly translated into charging requirements and transmitted to the proper departments.
- (c) The applicable core design parameters shall be clearly translated into operating information and transmitted to the proper departments.
- (d) The core design information necessary for other analyses shall be clearly identified, defined, and transmitted to the proper departments.

6.2.1.3 Evaluation

The restart criteria for core design examine the five aspects of the core design effort that dictate the level of confidence with which the reactor can be safely operated. These aspects include (1) establishing design responsibilities and requirements; (2) developing adequate methods and procedures for the determination of core control/shutdown system neutronics, including temperature coefficients and uncertainties; and (3) proper dissemination of design information. WSRC has several programs and documents, either in place or currently being developed, that collectively address the issues identified in the core design restart criteria.

Responsibility

Traditionally, core design functions at SRS have been shared by Savannah River Laboratory (SRL) and the Reactor Technology Department (RTD), a sub-group of Reactor Operations. The restart criterion on core design responsibility requires that Reactor Operations have the responsibility for concluding that the core safety evaluations are complete and reflect current plant conditions as well as the responsibility for defending the evaluations both within SRS and to disinterested third parties. A recommendation made by the Westinghouse Independent Safety Review Committee as a result of their evaluation of the SRS core design activities prompted WSRC to reevaluate the responsibilities for these activities. This reevaluation led to a restructuring of the core design functions and responsibilities within SRL and RTD.

The staff reviewed OPS-ROD-890205, "Core Safety Evaluation" (Reference 2), and found that as a result of the restructuring, a Core Performance Team (CPT) was established within WSRC Reactor Engineering, a sub-group under WSRC Reactor Operations. The CPT is made up of the managers of the Core Physics and Core Thermal-Hydraulics groups and three experienced engineers from these groups.

The CPT is responsible for integration and implementation of the core design and safety analyses, which includes responsibility for design specifications, verification that fuel received is within specifications, verification that core conditions lie within core design and safety analysis criteria, operating limits, physics and thermal-hydraulic data for operational assessment and operational data for core design and safety analyses. The CPT is also responsible for core performance analysis which includes estimated critical position predictions, shutdown reactivity predictions, control rod worth curves, operational anomaly analysis and observed performance summaries.

The responsibility criterion also requires that Reactor Operations establish the necessary interface agreements with the SRL for needed support. The staff found that these interface agreements have been defined in SCS-RGP-89-0020, "Centralization of Reactor Safety Responsibility" (Reference 3). These agreements establish the responsibilities for the flow and exchange of various information that is needed by the various organizations to accomplish the core design and safety analysis.

The staff concludes that the organizational changes recently implemented by WSRC provide adequate alignment of responsibilities and interface channels to satisfy the restart criterion for core design responsibility.

Design Requirements

Several of the General Design Criteria (GDC) defined in Appendix A of 10 CFR 50 (Reference 4) applied directly to the design of the reactor core. These GDC requirements are captured in the restart criteria under design requirements, item (2) of the criteria. They ensure that basic design considerations needed for safe reactor operation have been incorporated into the reactor design.

Limits on the power distribution of the core and margins to these limits must be established to preclude fission product release from the fuel due to fuel and cladding failure. In pressurized water reactors (PWRs) the ultimate limit is the limit on the departure from nucleate boiling ratio (DNBR), which quantifies how close the core is to experiencing fuel melting. Inherent to the DNBR determination are core power distribution parameters such as assembly average powers and hot channel factors (HCFs). Since these parameters help make up the DNBR, limits placed on the DNBR can be translated into limits on these power parameters.

The staff reviewed DPSTM-110, "Technical, Transient Protection and Confinement Protection Limits For SRP Reactors" (Reference 5), which described a limit for SRS reactors similar to the DNBR, called the nominal burnout safety factor (BOSF), that quantifies the likelihood of fuel melting due to boiling. Like the DNBR, the BOSF is made up of core power parameters so that limits on the BOSF at various plant conditions translate into limits on the core parameters such as HCFs. DPSTM-110 provides a discussion of the BOSF and how it is used to develop operating limits. BOSF limits may not be limiting under accident conditions that may correspond to different power shifts and thermal hydraulic conditions. A discussion providing an account of the uncertainties associated with the determination of the BOSF is given in DPSTM-22(H), "Hydraulic Design of the Mark 22 Charge" (Reference 6), and examples of the calculation and its derivation can be found in Volume 1 of DPSTP-RE-1.1, "Reactor Engineering Calculations" (Reference 7).

The staff found in DPSTM-110 that there are three types of operating limits determined for SRS reactors to control fission product release. They are the technical limit, the transient protection limit, and the confinement protection limit. The technical limit protects against fuel and cladding failure during normal operation and the transient protection limit during reactor transients terminated by reactor scrams. In both of these limits, the BOSF is used in the derivation indicating that acceptable values for power parameters must be determined for normal operating conditions and postulated transients that can be terminated by reactor scram.

During one design basis accident (DBA), transients may occur that are outside the scope of the transient protection limit analyses. Early in the DBA transient, a combination of event and conditions may create flow instabilities that could lead to fuel failures. The extent to which this situation is threatening is related to the operating power of the reactor. The power limits program discussed in section 11.3 of this SER examines the DBA and establishes the maximum allowable reactor power and applicable margins required to preclude fuel and target damage as a result of the DBA.

The staff has found that fuel and target design limits, in the form of power distribution limits and reactor power limits that maintain their respective integrity for both anticipated and postulated transients, are determined for the SRS reactor cores. Under the Power Limits (Section 11.3 of this SER) and SAR Chapter 15 analyses (Section 11.1 of this SER), reactor operating limits are being established to ensure fuel and target integrity is preserved.

The staff has reviewed several documents discussing the nuclear feedback of the core. The feedback, which is characterized by the temperature coefficient, can be separated into two components for the purpose of this discussion. One component is the prompt coefficient, which combines the effects of the fuel, coolant and target, and responds within 0.5 second of a change. The other component is the delayed coefficient, which is the effect of the moderator. The delayed coefficient begins to respond within about 1.5 seconds.

The requirement is that the feedback acts to compensate for positive reactivity insertions. Initial determinations of the temperature coefficients were reported in DPST-85-542, "Mark 22 Temperature Coefficients" (Reference 8). These determinations indicated that both the prompt and the delayed coefficients are negative at all times during core life. Later, DPST-87-237, "Coolant Temperature Coefficients In The NPR" (Reference 9), considered some information from a document detailing the results of some temperature coefficient measurements from 1962 (Reference 10). This information led the contractor to discover an additional phenomenon occurring in the fuel that made the prompt coefficient less negative than originally determined. The phenomenon involved expansion of the fuel and target tubes in response to temperature changes. Additionally, differences between measured values of the prompt coefficient given in DPST-62-419, "Mark VIB Temperature Coefficients" (Reference 10), and values calculated for fuel used in DPST-62-419 create a bias that has not been resolved. To be conservative, this bias is applied to the calculated prompt coefficient making it near zero early in life.

As a result of these actions, DPST-87-270, "Revised Mark 22 Coolant Temperature Coefficients" (Reference 11) reported a revised set of prompt coefficients that gave a value of zero for the prompt coefficient early in core life. DPST-88-956, "Limiting Values of Temperature Coefficients In Mark 22 Charges" (Reference 12), discussed the application of uncertainties to the calculated coefficients and gave a limiting value for the prompt coefficient that was slightly positive. All determinations of the delayed coefficient indicated that it was always significantly negative.

The fastest potential positive reactivity transient postulated for SRS cores is the startup accident. In this transient, groups of control rods are unintentionally withdrawn from the reactor at no power, or very low power conditions. This results in the creation of very short reactor periods before any temperature changes occur. WSRC examined the consequences of the startup accident with all but the least restrictive safety feature disabled and found that even with the limiting value of the prompt coefficient given in DPST-88-956, the transient would be safely terminated by safety rod scram. This analysis will be formally discussed in Section 11.1 of this report and will be part of the Chapter 15 analyses in the Safety Analysis Report (SAR).

Based on the reviews of Reference 8 through Reference 12 and of the preliminary analysis of the startup accident, the staff concludes that WSRC has adequately demonstrated that the net nuclear feedback inherent to SRS cores will compensate (fully at full-power and coupled with the safety rod scram at start-up) for the fastest positive reactivity transients postulated for the cores.

Power oscillations can be the result of two mechanisms during normal operation, xenon oscillation and a severely negative temperature coefficient. The staff reviewed DPST-87-1183, DPSTM-TP-2 and DPSTS-105, "105 Technical Specifications" (References 13, 14 and 15) and found that provisions are made during the design and operation of the core to reasonably ensure that power oscillations will not occur. During the design stage, DPSP-87-1183 and DPSTM-TP-2 invoke limits on how negative the temperature coefficient for the

core can be to preclude oscillations during normal operation. DPSTS-105 establishes technical specifications on axial and radial power distributions to help prevent the onset of xenon oscillation and terminate (via reactor scram) undetected or uncontrollable xenon transients before they threaten fuel design limits.

The staff found that in 1968 a study was done by SRL to determine xenon oscillation dependence on various reactor parameters (Reference 16). This study indicated that xenon oscillation depends primarily on the neutron flux level, spectrum, and shape; the neutron migration area; reactor dimensions; saturation xenon worth; and the temperature coefficient of reactivity. Of these parameters, only the neutron flux level and shape and the temperature coefficient were found to represent viable design variables.

The staff also found that experiments were done at SRS in 1970 to develop an understanding of the xenon oscillation behavior in SRS cores. DP-1214, "Xenon Oscillation Experiments" (Reference 17), and DP-1251, "Analysis of Xenon Oscillation Experiments" (Reference 18) report the results and findings from those experiments. The experiments indicated that although undamped xenon oscillations were possible in SRS cores (the most severe damping factor reported being 0.8), the period of oscillation for the cores was greater than 24 hours. This long period of oscillation provides ample time for the discovery and correction of the transients long before they threaten the technical specification limits on axial and radial power distribution.

The staff also reviewed RTR-2595 and DPSOL-105-1158, "Xenon Oscillation Control" (Reference 19), and found instructions for suppressing power oscillations should they occur during normal operation. Any power oscillation can be suppressed by inserting the safety rods and shutting the reactor down.

The staff concludes that WSRC has demonstrated that the SRS reactor cores are designed to minimize the potential for power oscillations during normal operation and that, under conditions where they do occur, technical specifications and operating procedures exist to control and suppress them.

Two independent shutdown systems are in place for each SRS reactor core. The staff reviewed Chapter 4 of the SAR and determined that the core is designed to be shutdown and maintained subcritical with either the combination of safety rods and control rods or with either of two trains of the Supplementary Safety System (SSS).

The 105 Technical Specifications specify that the safety rods, which are cadmium rods, must be capable of falling into the core via gravity and overcoming the reactivity contained in the temperature defect to render the core subcritical. Concurrently, upon a scram, the cadmium rods that are a part of each control rod septifoil will be driven into the core at high speed to provide negative reactivity compensation for the positive reactivity insertion that will occur when the xenon in the core has decayed away.

The specifications also state that two trains of the SSS be operable. The SSS injects the soluble neutron absorber, gadolinium nitrate into the core via six axially orificed injection tubes near the center of the core called sparjets. One train injects through one set of three sparjets and three pump inlets and the other through the other three sparjets and the other three pump inlets. The absorption capacity of one train of the SSS poison is sufficient to render and maintain the core subcritical and compensate for both xenon and temperature defects.

The major reactivity contributors in the core are the various isotopes in the fuel and targets, the coolant, the moderator, the control and safety rods, the axial and radial power distributions, and the gadolinium nitrate from the SSS system when injected. The staff reviewed DPSTM-500, "JOSHUA Manual" (Reference 20), and determined that the WSRC neutronics analysis of the core models all of these contributors at the lattice physics level with the exception of the power distributions (lattice physics calculations are two-dimensional infinite lattice models).

For finite reactor analysis, much of the contributors' behaviors are lumped into few-group nuclear constants that are numerically fitted to values of various core parameters such as exposure, uranium and lithium concentrations, fuel temperature, coolant and moderator temperatures and densities and gadolinium concentrations in the moderator. The finite reactor analysis provides an examination of the reactivity effects of the power distributions since it models the radial and axial assembly distributions in the core.

The staff acknowledges that many cycles of operating histories have provided an opportunity for good understanding of the reactivity contributors. The staff is also familiar with the scope and purpose of the power ascension testing (PAT) discussed in Section 8.3 of this report. Measurements of reactivity parameters, such as power coefficients, control rod worths, xenon worths and critical rod positions, will provide necessary benchmark data for verifying calculations and enhancing operating information.

The staff concludes that with the analysis capability at SRS, information provided by the PAT and the operating history data, WSRC can demonstrate adequate understanding and predictability of the core reactivity.

Design Methods

The primary core analysis tool at SRS is the Generalized Lattice Analysis Subsystem or GLASS code (Reference 21). GLASS is a two-dimensional, multi-group, integral transport theory, lattice physics code that models hexagonal spacings like those found in the SRS cores. GLASS can model up to 84 neutron energy groups and 200 spatial subregions. At SRS, GLASS is used to determine control rod worths, safety rod worths, temperature coefficients, and time dependent fuel and target isotopics. GLASS is also the primary generator of few group cross section and nuclear constant data for the finite reactor diffusion theory computer codes such as GRIMHX (Reference 22).

The staff reviewed the discussions pertaining to the theory and approximations used in the GLASS computer code modules contained in Volumes 4 and 4.1 of the JOSHUA manual. WSRC staff indicated that even though the JOSHUA manual was published in the early 1970s, it is still representative of many of the computer codes presently being used for analyses, particularly GLASS. The staff found the descriptions offered in these manuals are complete and provide enough information to conclude that methods used in GLASS are comparable to methods used in industry standard codes for similar applications such as THERMOS (Reference 23) based codes. The staff also acknowledges that the primary developer of GLASS was also a primary developer of the THERMOS code.

For the remainder of the reactor analysis codes, the staff reviewed the certification plan and the verification and validation plan for the reactor analysis codes developed by WSRC (Reference 24 and Reference 1). These plans are discussed in Section 5.8 of this SER, and will ultimately provide verification and validation for the reactor physics codes AA3, GILDA, GRIMHX, ICG, GLASS, JASON, MARCO, HMTABLE, PORAD, SHIELD, WIGGLE and TRIMHX. The plans identify what is needed to achieve certification and verification, the timetable and the resources needed to support these long-term efforts. The action matrix proposed in Reference 30 includes consideration of basics such as user's manuals and configuration control, descriptions of applicable theory, the development of tests and experiments for the purpose of validation, and the development and execution of benchmarks.

The staff acknowledges the effort outlined in Section 5.8 of this report and concludes that execution of the program offered in Section 5.8 will satisfy the restart criterion for all of the reactor physics computer codes.

The analysis and calculational methods necessary to perform the core design and analysis are given in the "Physics Charge Design Manual (Reference 13). The staff reviewed Chapter 6 of "Physics Charge Design Manual" and found detailed instructions for performing the core design analyses and calculations. The manual is formatted well and provides guidance for complete development of the core design.

Chapter 6 provides instructions for target selection, temperature coefficient calculations, coolant void coefficient calculations, control rod worths calculations, exposure and burnup calculations and power map preparation. It also gives instructions for preparing the charge design report data. This data includes fuel content determinations, control rod group reactivities, core average control rod worths, margins of control, migration areas, power distributions and lithium exposure calculations. The staff also reviewed the charge design technical manual DPSTM-TP-2 (Reference 14). This document was found to provide discussions about the bases and theory associated with the core design effort.

The staff concludes that the analysis and calculational methods presented in DPSP-87-1183 and the related discussions offered in DPSTM-TP-2 are comparable in scope to those found in commercial settings.

The uncertainties associated with the core design are generated primarily by two mechanisms: uncertainties generated by the computer analyses and uncertainties in fuel and target manufacturing. The computer code uncertainties will be addressed in the computer code verification and validation effort.

Manufacturing deviations from nominal should be accounted for in core design and safety analyses. The staff has reviewed WSRC-TR-90-42-014, "Uncertainties In Nuclear Test Gauge and Fuel Distribution Analyzer Measurements" (Reference 25), and OPS-RMT-900069, "Analysis of Fuel Distribution Analyzer Error Limits" (Reference 26), and found that they provide information quantifying the uncertainties that result from component manufacturing. The staff has reviewed the procedures for performing analyses and determined that they require these uncertainties to be considered in the analyses.

The staff concludes that the computer code verification effort and the procedural requirements for consideration of manufacturing uncertainties adequately satisfy this criterion.

Design Procedures

The staff found that the administrative and technical procedures for generating the core design are given in DPSP-87-1183. The staff also recognizes organizational changes described in SCS-RPG-89-0020 (Reference 27) and interface agreements previously discussed in this section of the SER that provide administrative control over the design.

DPSP-87-1183 identifies responsibilities and design constraints, specifies the necessary documentation, provides quality assurance guidance and provides analysis and calculational methodology for the core design effort. The staff has reviewed this document and found it to provide adequate guidance in each of these areas for acceptable core design.

Chapter 3 of DPSP-87-1183 contains the set of design constraints that are required for core design. These constraints include applicable technical specifications and standards and other requirements such as reactivity requirements, material requirements and core loading considerations.

Documentation of the core design is discussed in Chapter 4 of DPSP-87-1183. This chapter provides discussion and guidance for developing the Target Specification Memorandum, the Power Map Memorandum, Charge Limits Information Memorandum, the Physics Charge Design Report and Monthly Reports addressing the core design activities.

Quality assurance is accomplished through a number of reviews and checksheets in accordance with Section 8 of the Reactor Technology Standard Practice Manual (Reference 28).

In-house reviews are conducted for the activities associated with each document generated by the core design effort. These reviews are conducted by in-house engineers not associated with the given aspect of the design being reviewed and are meant to be independent reviews. Design inputs and outputs are reviewed and evaluated against the design requirements and constraints. Similar reviews are also conducted by SRL engineers as another independent check of the core design.

Charge design verification data sheets are completed by the engineer and signed by the core designer, the in-house reviewer, and an SRL reviewer. QA checksheets are also completed by the reviewer and signed. The QA checksheet requires signoffs on items such as reactivity predictions, minimum margin of control, lattice design, reactivity parameters, temperature coefficients, migration areas, fuel burnup, hydraulic factors, and reactor relative power maps.

Core Design Logbooks are maintained by the designer to provide an up-to-date source of information for the core design. The logbook contains a list of all the documents that pertain to a given design. The logbook is reviewed by both independent reviewers to ensure that adequate design controls have been realized.

The actions discussed above, document the core design effort in sufficient detail to establish the credibility of the executions of the design methods and procedures.

A match list checksheet (the match list is used to track assembly fabrication) is also prepared by the designer. This checklist requires review of the fuel and target specifications and requires the reviewer to indicate whether critical parameters are within specification. These parameters include number of assemblies per gang, core average Uranium-235, assembly Uranium-235, fractional fuel content, maximum Uranium-235 isotopic content range, maximum Uranium-236 -to- Uranium-235 ratio, assembly total buckling, region average Lithium-6 content, assembly Lithium-6 content and the disposition of nonconformance reports (NCRs) on out-of-specification parameters.

Design Products

The staff has reviewed the discussion on transmittals in Chapter 4 of DPSP-87-1183 and the charge specification document (Reference 29) and found that the necessary fabrications requirements are being transmitted to the applicable manufacturing departments. The specifications established in the charge specification document include average Uranium-235 content per assembly, individual assembly Uranium-235 content, axial variation of Uranium-235, maximum fuel density analyzer ratio, minimum Uranium-236 -to- Uranium-235 ratio and Lithium-6 contents and ratios.

The staff found that core loading information is transmitted to the component handling organization where it is developed into a set of information set used by the component handling equipment for properly loading the core.

The staff reviewed WSRC-RP-90-178, Reactor Analysis Responsibilities (Reference 30) and found that it established and outlined formal customer-supplier interface agreements for a core design information exchange between SRL and RE (operations). Charge limits information consisting of control rod worths, temperature coefficients, and core exposure data are generated during the core design effort and transmitted to Operations for use during reactor operation.

6.2.1.4 Conclusion

The staff has reviewed the WSRC programs that are proposed and in place for core design and has determined that the program does satisfy the restart criteria.

6.2.1.5 References

1. WSRC, WSRC-RP-89-1249, "Verification and Validation Plan For Reactor Analysis Computer Codes."
2. WSRC, OPS-ROD-890205, "Core Safety Evaluation" (WISRC-23).
3. WSRC, WSRC-TM-90-4, "LOCA FI Limits Methodology (U)," (draft).
4. Title 10, Code of Federal Regulations, Part 50 (10 CFR 50).
5. WSRC, DPSTM-110, "Technical, Transient Protection and Confinement Protection Limits For SRP Reactors."
6. WSRC, DPSTM-22(H), "Hydraulic Design of the Mark 22 Charge."
7. WSRC, DPSTP-RE-1.1, "Reactor Engineering Calculations, Volume 1."
8. WSRC, DPST-85-542, "Mark 22 Temperature Coefficients."
9. WSRC, DPST-87-237, "Coolant Temperature Coefficients In The NPR."
10. WSRC, DPST-62-419, "Mark VIB Temperature Coefficients."
11. WSRC, DPST-87-270, "Revised Mark 22 Coolant Temperature Coefficients."
12. WSRC, DPST-88-956, "Limiting Values of Temperature Coefficients In Mark 22 Charges."
13. WSRC, DPSP-87-1183, "Physics Charge Design, Volume 1."
14. WSRC, DPSTM-TP-2, "Charge Design (Physics) Volume 1."
15. WSRC, D{STS-105, "105 Technical Specifications."
16. WSRV, DPST-68-372, "Xenon Oscillation Dependence On Reactor Parameters."
17. WSRC, DP-1214, "Xenon Oscillation Experiments."

18. WSRC, DP-1251, "Analysis Of Xenon Oscillation Experiments."
19. WSRC, DPSOL-105-1158, "Xenon Oscillation Control."
20. WSRC, DPSTM-500, "JOSHUA Manual."
21. WSRC, DPSTM-89-100-1, "GLASS User's Manual."
22. WSRC, DPSTM-86-100-1, "JASON Reactor Design Code."
23. Brookhaven National Laboratory, BNL-5826, "THERMOS, A Thermalization Transport Code For Reactor Lattice Calculations."
24. WSRC, WSRC-TR-90-26, "Certification Plan For Reactor Analysis Computer Codes."
25. WSRC, WSRC-TR-90-42-014, "Uncertainties In Nuclear Test Gauge and Fuel Distribution Analyzer Measurements."
26. WSRC, OPS-RMT-900069, "Analysis of Fuel Distribution Analyzer Error Limits."
27. WSRC, SCS-RPG-89-0020, "Centralization of Reactor Safety Responsibility."
28. WSRC, DPSP-87-1211-2, "Reactor Program Standard Practice Manual."
29. WSRC, WSRC-RP-89-1289, "K-14 Mark 22 Charge Specification."
30. WSRC, WSRC-RP-90-178, "Reactor Analysis Responsibilities."

6.2.2 Control Rod Drive System Reliability

6.2.2.1 Background

The Control Rod Drive and Rod Reversal System (CRDS) is the means by which reactor power and neutron flux are controlled from sub-critical to full power, including shutting down the reactor when required. The design basis of the CRDS is as follows: (1) to regulate reactor power, (2) to adjust the neutron flux for optimum power and safety, (3) to produce certain nuclear materials in the control rods, (4) to shut down the reactor under normal conditions, and (5) to maintain the reactor in a shutdown condition after a safety rod scram. The CRDS consists of a number of rod clusters containing both axial and radial flux controlling rods. The radial (full) and axial (partial) flux control rods are positioned via a combination of N (Full) and O (Partial) rod demand position units, T-Amplifier sensing and Control units, and Control Rod Drive motor packages.

A positioning demand signal is supplied to the rod drive motor synchronizer from the N or O unit. The rod drive synchronizer compares the positioning signal to the actual position and develops an error signal proportional to the difference. The T-Amplifier resolves this signal and other inputs to provide the appropriate rod drive motor control. The rod drive motors move the rods through a rack and pinion system to the required position in the reactor core. Upon initiation of a scram signal, the safety rods drop into the core, and control rods are driven into the core at an increased speed until all of the control rods are fully inserted into the core.

Due to spurious control rod movement, control rod drift, control rod mispositioning during reactor operations, and failure to identify unlatched control rods during reactor startup the reliability of the CRDS has come into question over the past several years, resulting in evaluations by DuPont, WSRC, and DOE.

During a reactor scram, the safety rods are required to drop into the core and completely shut down the reactor. However, after 1.5 hours, the reactor may return to criticality unless the control rods are inserted. Therefore, the reliability of the CRDS must be established in order to maintain the necessary margin of safety in the SRS reactor plants.

The T-Amplifiers were identified as a major CRDS reliability concern in the Reactor Issue Management Program-172 (Reference 1). These control modules had been in service since the beginning of operation at SRS (circa 1953). They were of a vacuum-tube design and increasingly difficult to maintain as spare parts became more difficult to obtain. The decision was made to replace the vacuum-tube type T-Amplifiers with new modular state-of-the-art models. Fifteen prototype T-Amplifiers were designed and constructed at SRS and installed in the P-Reactor in 1986. From 1986 until 1989, these prototype T-Amplifiers performed with only one failure. The data obtained from these prototypes was used in the final design of the new T-Amplifiers. The new T-Amplifiers have been installed in K-Reactor and are scheduled to be installed in each of the reactors prior to startup.

The new T-Amplifiers have themselves exhibited a history of problems. Initial shipments from the vendor were unacceptable, consisting of broken and damaged parts and missing and/or incomplete documentation. Initial bench tests designed to determine if T-Amplifiers responded with correct output signals and voltages based on designed voltage and signal input identified additional problems. The most significant of the problems were T-Amplifier fuse blowing and T-Amplifier relay contacts "welding" closed (excessive contact arcing). EED later identified the cause of the current surges to be the electrical "noise" that was generated mainly from control rod drive motor package synchronizers. A manufacturing problem, cracked diodes, resulted in a reliability concern regarding the new T-Amplifiers.

6.2.2.2 Restart Criteria

The criteria for determining the adequacy of the CRDS for restart are described below:

- (1) The procedure for latching the control rods to the actuator tower extension rods shall include an independent verification that the latch is engaged. An independent verification shall also be done at the first hold point on power ascension. (DB-3.4 of the 1988 Savannah River Site K-Reactor Restart Strategy) (Reference 2.)
- (2) Spurious movement of the control rods shall be investigated and documented. Spurious control rod movement shall be reported to the senior shift supervisor and the Reactor Technical Division.
- (3) Control rod system maintenance shall be done at the highest quality level defined in the SRS quality control manual.
- (4) Procedures shall be developed and implemented for trouble shooting that clearly define the allowable actions for trouble shooting. Results of trouble shooting are to be documented. When activities outside the procedure are required, work shall be stopped and the procedure shall be formally changed and concurred with by the cognizant reactor technical system engineer and reactor technical supervisor and approved by the senior shift supervisor.
- (5) Technical Specifications shall define operating limits, such as the minimum number of rods operable during start-up and operation, for the control rod system.
- (6) Maintenance and trouble shooting histories of the control rod system shall be analyzed to detect and correct defects.
- (7) The T-amplifiers (control center for the control rod drive motors) shall be replaced on a phased schedule with solid state equivalent amplifiers. The schedule for replacement shall be submitted for DOE approval.
- (8) Preventative maintenance shall be performed on the following CRDS components: latches, motor packages, relays, and N, O, and trim servo units.

- (9) Control computer programming shall be developed to collect the control rod gang demand data, the four high level flux monitor (HLFM) meter readings, and the 2 Log-N meter readings and to transfer these data to an off-line computer. (DB-3.5 of K-Reactor Restart Strategy.)

6.2.2.3 Evaluation

Restart Criterion 1

The staff reviewed DuPont Standard Operating Log (DPSOL) 105-1135A-PLK, Revision 2, "Control Rod Pattern and Latch Check-Reactor Startup" (Reference 3), (approved December 6, 1989). The data sheets for this procedure have provisions for two individuals to verify rod latching. In the General Limitations and Precautions section, Item 5 stipulates that the second person completes the check after the first person has completed the check. This administratively establishes independence of the two checks. The frequency of performance of this procedure is (1) within 8 hours prior to startup, and (2) within 4 hours after reaching low power. Based on this review, the DOE staff has determined that this restart criterion is met.

Restart Criterion 2

The staff reviewed DPSOL 105-1715-PLK, Revision 28, "Fault Correction and Record of Control Rod Malfunctions" (Reference 4) (approval date January 26, 1990). The "Purpose" section adequately defines under what circumstances this procedure is to be utilized. These circumstances are: (1) correcting T-Amplifier faults, (2) recording rod malfunctions, and (3) listing unwanted control rod motion during reactor startup, operation, or testing. Item 3 in the "General Limitations and Precaution" section requires notification of supervision prior to implementing this procedure due to the serious consequences that can occur during reactor operation. Data Sheet 1 is required to be completed for unwanted control rod motion occurrences. Step 21 of this data sheet requires that a copy of the data sheet be forwarded to Reactor Engineering. Reactor Engineering personnel have been assigned to 24-hour shift coverage ensuring review of this data sheet will occur in a timely manner. Based on this review, the DOE staff has determined that this restart criterion is met.

Restart Criterion 3

The staff reviewed the system classification requirements, the current maintenance procedures, and the Westinghouse Quality Assurance Manual (QAM) (Reference 5) for activities relating to maintenance. Per RTR-2631 (Reference 6), "Classification of Reactor Systems Important to Safety," Safety Class 3 section, paragraph (e), the control rod system is classified as a Safety Class 3 system relied upon to "introduce negative reactivity to achieve or maintain the reactor system in a less-than-critical state, but not needed to rapidly scram or shutdown the reactor." A reactor maintenance self-assessment, conducted to the recommendations of the Institute of Nuclear Power Operations (INPO 85-038, "Guidelines for the Conduct of Maintenance at Nuclear Power Stations" (Reference 7), identified the need to regiment the maintenance practices on reactor plant systems. The maintenance procedure upgrade program

is intended, in part, to require quality inspections and involvement in the reactor maintenance program. As a result of the self-assessment, Reactor Restart Division Administrative Manual RD 3.1, procedure number DM 3.01, "Work Control" (Reference 8), was developed to stipulate the requirements for the conduct of all maintenance on reactor plant systems, components, and structures. The requirement to develop work packages for all maintenance performed and the level of review that each work package receives (both on development and completion) provide confidence that the reactor maintenance program is acceptable. The QAM, chapter QAP 10-1, Section 3.0, "Responsibilities", delineates the various organizational responsibilities necessary to perform quality work on reactor systems. Among these are the requirement for all organizations to establish the procedural controls to ensure performance of independent inspections to notify the Cognizant Quality Function (CQF) of the need for performance of independent inspections at specified hold points or witness points in working documents. Section 4.1 of QAP 10-1, "Specifying Inspections", identifies the requirements for the CQF to perform various independent inspections. Quality Hold Points are identified and defined in this section along with other types of inspections. Based on this review, the DOE staff has determined that this restart criterion is met.

Restart Criterion 4

The reactor maintenance self-assessment noted that there were no formal maintenance troubleshooting procedures. INPO 85-038, "Chapter VII, "Control of Maintenance Activities," section B, recommends that "troubleshooting should be controlled to prevent unplanned repairs and unauthorized modifications." As a result of the reactor maintenance self-assessment DPSOL 1-70420, "Troubleshooting and Minor Maintenance," Revision 0, October 1, 1990, (Reference 9), was developed to provide a formal, documented and controlled approach to performing troubleshooting maintenance. Specific attributes encompassed by the troubleshooting program are (1) adequately describe the troubleshooting actions, (2) document the results of the troubleshooting activities, and (3) prescribe the necessary actions to be taken if and when activities outside of the prescribed plan were encountered. The WSRC troubleshooting procedure requires work packages to be assembled and adequately reviewed by cognizant engineering personnel prior to beginning the trouble shooting evolution. Provisions are also included in this procedure to describe the actions to be taken if activities outside of the procedures are necessary. Based on this review, the DOE staff has determined that this restart criterion is met.

Restart Criterion 5

The staff reviewed the Technical Specifications (TS) for Savannah River Plant Operations (WSRC-TS-10003, Revision 0, dated December 18, 1990) (Reference 10). Technical Specifications examined were:

- 3.1.3 Reactivity Requirements Shutdown
- 3.2.1 Full Length Control Rod Positioning
- 3.2.2 Partial Length Control Rod Positioning
- 3.8.1 Control Rod Actuator Power Supply

It has been determined that the current Technical Specifications adequately address this restart criterion.

Restart Criterion 6

The staff reviewed INPO 85-031, Guidelines for the Conduct of Technical Support Activities at Nuclear Power Stations (Reference 11), INPO 85-033, TS 413, "Use of System Engineers" (Reference 12), and two Inter-Office Memoranda (References 13 and 14) issued by the WSRC Reactor System Engineering Manager defining the duties and responsibilities of WSRC system engineers (OPS-RSE-900433, dated March 13, 1990, and OPS-RSE-900438, dated March 14, 1990). The INPO documents recommend that the System Engineering and the Technical Support organizations periodically evaluate system performance to improve system efficiency and reliability, to trend important system parameters, and to initiate corrective action prior to equipment failure. The WSRC Reactor System Engineering memoranda concerning System Engineering duties and responsibilities closely parallel the attributes of the INPO documents and direct the System Engineers to obtain and review system data and maintenance records. Engineering personnel are directed to review maintenance work orders, perform regular reviews of operating logs, review Event Reports, and to coordinate troubleshooting activities on systems and components along with other engineering duties. Although these memoranda capture the INPO performance criteria, they are not formal plant procedures. Pending issuance of formal procedures, compliance with this restart criterion is an open item.

Restart Criterion 7

WSRC has replaced the vacuum tube T-Amplifiers with solid state models. However, there have been problems with the new T-Amplifiers. This restart criterion will remain an open item until all problems are resolved, and the new T-Amplifiers are demonstrated to be acceptable for their application.

Restart Criterion 8

DPSOL 9-10900 (Reference 15) is a general instruction requiring a detailed work package to be issued to perform the indicated maintenance. Several types of PMs are identified in this base document, including motor overhaul and visual inspection. This DPSOL was used in performing the PM for the Full and Partial rod synchronizers (N, O, and Trim units). The procedure was last performed in June 1990 for the N and O units and in March 1990 for the Trim units. During performance of CRDS checkout procedures, it was determined that there were several conflicts between the field wiring and the design requirements for the N and O units. As a result of these and other unit related problems, WSRC developed specific overhaul procedures for the N and O units. WSRC procedures SOP 5-10529 (Reference 16) and 5-10531 (Reference 17) were revised to provide detailed instructions for the overhaul wiring check, and component inspections for the N and O units. Spare N and O units were completely overhauled per these procedures.

DPSOL 1-82222, "Actuator Synchro System Checkout" (Reference 18), is used for checking proper operation of the control rod drive system synchronizer units. This procedure provides documentation of the electrical signals generated in

the CRDS. The control rod position demand signal is compared to the actual control rod position and an error signal voltage is generated through the T-Amplifier, creating rod movement. This procedure is a prerequisite to the performance of WSRC-OX-89-27-001, Test Authorization 1-2297, Special Procedure (SP) 2456 (Reference 19). During preparation for performance of SP 2456, DPSOL 1-82222 was initiated and was stopped due to failure of the N and O units to function correctly. As a result of this failure, the N and O units were overhauled and rewired as identified in the above paragraph. The K-Reactor N and O units are now undergoing another rework process to correct a lug trimming error.

Testing of the control rod motor packages is included in performance of DPSOL 1-82222. All of the motor packages have been rebuilt per-DPSOL 1-74002, "A-Motor Package Inspection and Maintenance" (Reference 20). This DPSOL is a general PM procedure for overhaul, lubrication, and setting of limit switches. Due to problems with the new T-Amplifiers, the motor packages are being tested for faulty synchronizers, as described in Reactor Technical Memorandum (RTM) 5055 (Reference 21).

DPSOL 1-82204, "Bench Test of T-Amplifier" (Reference 22) determines the operability of the T-Amplifier. This procedure is performed on the T-Amplifier bench tester. All of the new solid state T-Amplifiers will be subjected to this test prior to reactor restart.

Control rod latches were inspected and cleaned in May 1990 during preparations for performance of SP 2456. There are no administrative requirements to ensure periodic PM of the latches.

Pending completion of the required PM on CRDS components and the establishment of an adequate PM program, compliance with this restart criterion is an open item.

Restart Criterion 9

The staff assessment consisted of observing the performance of the computer hardware and software in the Reactor Engineering Program Development Center (PDC), verifying the schedule to support installation of the system in K-Reactor, and verifying operator training requirements. The observation consisted of a DOE staff walkdown with the development engineer of the hardware installation in the PDC and a demonstration of the software. The new software program gathers 3 parameters from the control computers every 15 seconds and transfer this data every 10 minutes to an off-line computer located in the reactor area computer room. The data include the control rod gang demand data, the 4 High Level Flux Monitors, and both log-N channels. The observed demonstration provided sufficient information as to the ability of this program to achieve the desired results. On-the-job training will be used to train the reactor operation personnel in the use of this program. The current schedule indicates that the installation will support K-Reactor restart. Functional testing of this program will require support of the CRDS and will be scheduled after performance of SP 2456. Based on this review, the DOE staff has determined that this restart criterion is met.

6.2.2.4 Conclusion

Although progress is being made toward satisfying the restart criteria, the new T-Amplifiers continue to have problems. Resolution of the open items identified above will be documented in an SER Supplement.

6.2.2.5 References

1. WSRC, "Westinghouse Savannah River Company Reactor Issues Management Program (RIMP)-172," September 27, 1989.
2. WSRC, "Savannah River Plant K-Reactor Restart Strategy," November 1988.
3. DuPont Standard Operating Log (DPSOL) 105-1135A-PLK, "Control Rod Pattern and Latch Check-Reactor Startup," Rev. 2, December 6, 1989.
4. DuPont Standard Operating Log (DPSOL) 105-1715-PLK, "Fault Correction and Record of Control Rod Malfunctions," Rev. 28, January 26, 1990.
5. WSRC, Westinghouse Quality Assurance Manual (QAM), WSRC-1Q, Rev. 5, November 1, 1990.
6. WSRC, Letter E. Noim to F. Beranek, Reactor Technical Report (RTR)-2631, "Classification of Reactor Systems Important to Safety," April 7, 1989.
7. Institute for Nuclear Power Operations, INPO 85-038, "Guidelines for the Conduct of Maintenance at Nuclear Power Stations," Rev. 1, December 1988.
8. WSRC, Reactor Restart Division Administrative Manual RD 3.1, Reactor Work Control Manual, Procedure Number DMP 3.01, "Work Control," Rev. 3, January 25, 1991.
9. DuPont Standard Operating Log (DPSOL) DPSOL 1-70420, "Troubleshooting and Minor Maintenance," Rev. 0, October 1, 1990.
10. WSRC, WSRC-TS-10003, "Technical Specifications (TS) for Savannah River Plant Operations," Rev. 0, December 18, 1990.
11. Institute of Nuclear Power Operations, INPO 85-031, "Guidelines for Technical Support Activities at Nuclear Power Stations," Rev. 1, September 1988.
12. _____, INPO 85-033, TS-413, "Use of System Engineers," September 1985.
13. WSRC Memorandum OPS-RSE-900433, S. D. Curry to All RSE Personnel, "Reactor System Engineering Department Organization Changes," March 13, 1990.

14. WSRC Memorandum OPS-RSE-900438, S. D. Curry to Distribution, "Desk Top Instruction On Investigations of Root Cause for Equipment Failure," March 14, 1990.
15. DuPont Standard Operating Log (DPSOL) 9-10900, "Preventive Maintenance Job Plan," Rev. 7, May 7, 1987.
16. WSRC, Standard Operating Procedure (SOP) 5-10529, Manual Y7.1, "O-Unit Repair, EP 105-159.5," Rev. 6, August 3, 1990.
17. WSRC, Standard Operating Procedure (SOP) 5-10531, Manual Y7.1, "N-Unit Repair, EP 105-159.4," Rev. 5, August 3, 1990.
18. DuPont Standard Operating Log (DPSOL) 1-82222, "Actuator Synchro System Checkout," Rev. 7, April 20, 1990,
19. WSRC, WSRC-OX-89-27-001, Test Authorization 1-2297, "Check Out for Solid State T-Amplifiers and Control Rod System 105 K," Special Procedure (SP) 2456, Rev. 2, July 6, 1990.
20. DuPont Standard Operating Log (DPSOL) 1-74002, "A-Motor Package Inspection Maintenance," Rev. 6, January 14, 1991.
21. WSRC Reactor Technical Memorandum (RTM)-5055, J. L. Hendrix to E. P. Rahe, "A-Motor Package 36X Synchro Replacement," July 13, 1990.
22. DuPont Standard Operating Log (DPSOL) 1-82204, "Bench Test of T-Amplifier," Rev. 9, July 28, 1977.

6.2.3 Fuel Quality

6.2.3.1 Background

The fuel and target assemblies that are irradiated in the Savannah River reactors are manufactured on the site. Mark 22 assemblies, which contain two fuel and two target tubes, and Mark 60B assemblies, which are single-target tubes are fabricated in what is known as the 300-Area. Individual fuel and target tubes are manufactured to within specified tolerances. Fuel and target tubes are then selected to be combined to make assemblies that satisfy the manufacturing requirements through what is called the matching process. The matching process produces documentation (referred to as "match lists") that indicates which fuel and target tubes make up a given assembly.

Due to errors found between the match lists used for tracking fuel assemblies and the assemblies actually sent to the reactor areas, DOE issued a Stop Work Order for fuel and target fabrication (300-Area) in the fall of 1988. In order to have the Stop Work Order lifted, DOE required DuPont/WSRC to identify and correct the system failures that had allowed the errors to occur. In response to DOE's request, WSRC developed a Product Certification Program and began implementing it in the summer of 1989. This program addresses issues such as manufacturing uncertainties, material quality, traceability, testing and inspection, and independent review of manufacturing area products. DOE evaluated the Product Certification Program and lifted the Stop Work Order in the fall of 1989. The purpose of this evaluation is to assure that fuel and target fabrication processes are performed within established technical specifications, QA, and accountability requirements.

6.2.3.2 Restart Criteria

The criteria for determining the adequacy of the SRS fuel quality program for restart are described below.

(1) Fabrication Requirements

- (a) Fabrication requirements shall be specified which ensure that core physics and thermal-hydraulic design and safety analysis are satisfied.
- (b) The materials used shall be of a quality that meets or exceeds those materials specified in the fabrication requirements.
- (c) The physical dimensions of component parts shall conform to the fabrication requirements to within specified tolerances.
- (d) The physical dimensions of the assembled components shall conform to the fabrication requirements to within specified tolerances.
- (e) The material distributions within the assembled components shall conform to the fabrication requirements to within specified tolerances.

(2) Fabrication Processes

- (a) The casting processes used in fabrication of the core components shall be specified in sufficient detail to establish that the desired product quality can be achieved.
- (b) The extrusion processes used in fabrication of the core components shall be specified in sufficient detail to establish that the desired product quality can be achieved.
- (c) The assembly processes used in fabrication of the core components shall be specified in sufficient detail to establish that the desired product quality can be achieved.
- (d) The finishing processes used in fabrication of the core components shall be specified in sufficient detail to establish that the desired product quality can be achieved.

(3) Fabrication Procedures

- (a) The administrative procedures used to fabricate the core components shall be sufficient to establish that all required interfaces and approvals reflect, or are comparable to, the current quality assurance standards.
- (b) The technical procedures used to fabricate the core components shall be sufficient to establish that all fabrication inputs, method exceptions, and outputs reflect, or are comparable to, the current approved quality assurance standards.
- (c) The procedure for documenting the fabrication effort shall be sufficient to establish the credibility of the execution of the fabrication methods and procedures.
- (d) The procedure for tracking components parts and assembled components provides a clear, unquestionable history of the fabrication effort.

(4) Fabrication Assessments

- (a) The uncertainties associated with the fabrication are clearly stated and evaluated, and are acceptable with respect to design specifications.
- (b) Required characterization testing and inspections of assemblies or their components shall be performed, and the components shall meet or exceed the fabrication requirements specified by the core design.
- (c) The documentation for the core components inventory shall be independently reviewed and shall provide sufficient administrative and technical detail to allow complete and permanent traceability.

6.2.3.3 Evaluation

The program that Westinghouse Savannah River Company (WSRC) has put into place to control fuel and target manufacturing quality is described in the 300 Area Product Certification (Reference 1). This program introduces the concept of "process data sheets" to provide a complete lifetime quality/accountability record for the manufactured components. There are four process data sheets associated with each Mark 22 assembly (References 2 through 5) and one process data sheet for each Mark 60B assembly (Reference 6). The data sheets are in the form of the 300 Area procedures and are controlled documents. The program also established certification procedures for both the Mark 22 (Reference 7) and Mark 60B (Reference 8) assemblies and two new match list procedures, one for the actual matching of fuel and target tubes (Reference 9) and the other for making changes to the match list (Reference 10). The program was fully implemented as of March 1990, and the K-14 core is a fully certified core under the new program.

Fabrication Requirements

In determining the extent to which the fabrication requirements are satisfied, the staff considered the four elements specified in Restart Criterion 1 (i.e., adequacy of fabrication specifications, assurance of material quality, demonstration of dimensional conformance, and verification of material distributions).

The staff found that the manufacturing organization establishes the adequacy of the fabrication specifications through verification that the specifications were taken from the proper Reactor Technology Memoranda (RTM); thus, making the assumption that the RTMs are products of an adequate design effort. This verification is documented on the certification documents that are associated with each assembly. The RTMs provide the manufacturing organization with uranium and lithium content information, fuel and target matching requirements, and applicable assembly and assembly component drawing information. The staff's evaluation of the RTM fabrication specifications is contained in SER Section 6.2.1.

The quality of material for fuel and target manufacturing is dictated by a set of standards called "Essential Material Specifications" (EMSs). The staff reviewed the process data sheets for each type of subassembly and found that the applicable EMSs are stated on the data sheets in the "Charge Makeup" section of the sheets and that verification of material quality is required before any of the manufacturing processes begin. This is accomplished by a reviewer (supervisor) comparing recorded values to the required values listed on the data sheets. Nonconformance reports (NCRs) are required for any material found outside of the EMSs, and the procedure requires that either a Reactor Materials Quality (RMQ) supervisor or a Reactor Materials Production supervisor disposition the NCR. The NCR number is recorded on the process data sheet associated with the nonconforming material.

In reviewing the process data sheets, the staff found a very rigorous dimensional examination and documentation program used as recorded on the data sheets for each subassembly and the final assembly. Dimension specifications

for the components at various stages of development have been taken directly from the assembly drawings and incorporated on the data sheets. The final inspection is documented under the heading "Tube Dimensional Inspection" on the data sheet. Actual measured values of component dimensions and asymmetries are required to be recorded adjacent to the listed specifications for comparison. A RMQ inspector independently reviews the inspection data and determines whether the component is to be accepted or rejected based upon the criteria listed on the data sheet. The status is indicated on the process data sheet. The RMQ inspector then signs and dates the data sheet.

The process data sheets provide for a documented evaluation of the material distributions within the subassemblies and assemblies. The staff found a fluoroscope inspection section, fuel distribution analyzer (FDA) section, a nuclear test gauge (NTG) section and a section for associated destructive examination. Each of these sections provide the desired manufacturing specifications and requires that the measured values be recorded adjacent to the applicable specifications for comparison. The accepted/rejected status of the component is required to be recorded on the data sheet. The sections of the sheet are then signed and dated.

Fabrication Processes

Requirements for the processes that are used to manufacture the subassemblies and assemblies are given in Restart Criterion 2. This criterion requires that the casting, extrusion, assembly and finishing processes be sufficient to deliver the desired product quality. The procedures associated with each of these processes for a given subassembly are listed in the reference section of each of the subassembly process data sheets. The staff found that the process data sheets ensured that each process provided products of desired quality by requiring that various parameters associated with each of the processes be determined, recorded and compared to desired values or ranges. NCRs are written for those products not satisfying the requirements, and the NCR number is recorded on the process data sheet in the appropriate section, with designated signatures and dates.

The staff found that for the casting process, the data sheets require documentation of uranium, lithium and aluminum weights, signatures and dates for the casting, and casting assay information including whether the casting is accepted or rejected. Rejected castings have NCRs written for them with the NCR number recorded on the data sheet for the material.

The data sheets require evaluation and documentation of various steps in the extrusion process. Operator signatures and dates are required to indicate the performance of pre-extrusion core machining. A pre-extrusion core inspection is required that examines lengths, diameters, wall thicknesses and angles, and inspects for surface voids and inclusions. Again the acceptance status is indicated on the data sheet; however, two RMQ inspector's signatures and the dates are required. Pre-extrusion billet assembly, welding, verification and cleaning are all documented with applicable information and signatures and dates on the product process data sheet. The data sheets require evaluation and documentation of coextrusion core machining, shuffler assay, core inspection, billet assembly, billet welding and billet verification; and

post-extrusion cleaning, blister inspection, drawing, straightening, degreasing and final machining. Each of these evaluations requires an acceptance determination to be made as a part of the evaluation, and designated signatures and dates are recorded.

The outer target process data sheet lists the procedures used to create a final assembly. This data sheet provides for evaluation and documentation of the assembly and finishing processes in addition to that for the outer target tube fabrication. This evaluation considers fuel and target tube selection, bottom fitting crimping verification, magneforming, assembly inspection and final verification. Once again, accepted or rejected status must be indicated for each evaluation, and designated signatures and dates are recorded.

Fabrication Procedures

Restart Criterion 3 examines the adequacy of the procedures used to manufacture the subassemblies and assemblies. In reviewing the procedures that resulted from the 300 Area Product Certification, the staff found that the process data sheets provide well-defined administrative controls for manufacturing. The sheets identify the purpose, responsibilities, technical procedures, required training, documentation controls, and the necessary interfaces and approvals to produce a certifiable product. Additionally, the certification procedures require review of the completed data sheets for the assemblies and checks that the requirements of the RTM have been satisfied, and that all required documentation is in place and complete.

The adequacy of the technical procedures are reflected through the information recorded on the process data sheets. The data sheets list the applicable technical procedures, and require indications through sign-offs that the procedures have been completed properly. The data sheets also require that detailed information from each manufacturing process be recorded, where applicable, providing a record of the outputs from the current process and the inputs to the next process. When processes produce products that are not acceptable, NCRs are written for the products, documented on the product's data sheet, and dispositioned by RMP and RMQ supervisors.

As indicated throughout this discussion, the process data sheets provide a very clear, concise, and complete documentation of the assemblies and of the subassemblies that are used in making given assemblies. Process information from initial material selection to final tube matching are contained on the data sheets with dates and signatures corresponding to evaluations of each process output. This documentation is supplemented by the certification procedure documents that provide the final review and verification of the assembly's quality.

Subassembly and assembly tracking is accomplished with the process data sheets, the certification procedures, and the match list procedures. The match list procedures provide guidance on match list preparation, revision, identification, control and distribution. These procedures provide rigorous checks to ensure that proper assembly requirement information is being used and that the matching corresponds to that information. Changes to the match list are controlled through the "Match List Change Request Form" procedure.

This procedure defines what can cause a match list change to be initiated, the steps necessary to generate the change request, how the request must be reviewed and approved, how the change request is identified, how the request is documented, and how the request is distributed.

The process data sheets, the certification procedure, and the match list all become part of the assembly's history record, and are retained for at least the life of the assembly.

Fabrication Assessments

The assessment requirements for the manufactured components are given in Restart Criterion 4. Uncertainties associated with fabrication are reflected in the measured values of process parameters recorded on the data sheets. The data sheets also require that the measuring device identification number be recorded adjacent to the measured value on the data sheet. Acceptable tolerances for each parameter are given on the data sheet with the nominal value of the parameter and are found adjacent to the measured value.

The process data sheets require a rigorous battery of tests and inspections for their adequate completion. The acceptable ranges for various parameters are provided on the sheets in locations adjacent to the space provided for the measured value to be recorded. A determination of the adequacy of the product is required for each process step, with NCRs generated for those products failing to meet the standards. Reviews and sign-offs by RMP operators are required for each process, with RMP and RMQ supervisor sign-offs required for certain intermediate steps and the final approval.

Independent review of the manufacturing documentation is accomplished through the certification documents. These documents require verification that RTM specifications have been satisfied, process data sheets have been completed and are in place for each subassembly, the final match list is complete, and all NCRs associated with the final assembly have been adequately resolved. This documentation is signed and dated by the preparer and the reviewing supervisor.

6.2.3.4 Conclusion

The staff has reviewed the WSRC program for fuel manufacturing and determined that the program satisfies the restart criteria for fuel quality.

6.2.3.5 References

1. WSRC, WSRC-RP-89-676, "300-Area Product Certification."
2. WSRC, SOP 300-3002 I, "Mark 22 Inner Target (AI) Process Data Sheets."
3. WSRC, SOP 321-3011 I, "Mark 22 Inner Fuel (BI) Process Data Sheets."
4. WSRC, SOP 321-3010 I, "Mark 22 Outer Fuel (BO) Process Data Sheets."

5. WSRC, SOP 300-3001 I, "Mark 22 Outer Target (A0)/Mark 22 Assembly Process Data Sheets."
6. WSRC, SOP 300-3000 I, "Mark 60B Process Data Sheets (U)."
7. WSRC, DPSOL 321-1455 I, "Certification of Mark 22 Fuel and Target Assemblies."
8. WSRC, DPSOL 320-3009 I, "Certification of and Shipping Mark 60B Target Subassemblies."
9. WSRC, DPSOL 321-2897C, "Matching Fuel and/or Target Tubes."
10. WSRC, DPSOL 321-2898C, "Match List Change Request Form."

6.3 Electrical Issues

6.3.1 Electrical Power Systems

6.3.1.1 Background

The production reactors at Savannah River were designed decades prior to the development of the industry standards that exist today. The reactors have been undergoing upgrading on a continuing basis since the original operations started in the early 1950s. For example, when the plant originally started operation, there was only one electric power line to the commercial electric grid; today, there are three. The evolution of the electric power system at SRS has not followed the industry standards that have been developed. This is due, in part, to the differences between a heavy water production reactor and a light water utilization reactor that generates steam and electricity. Therefore, the production reactors do not currently comply with today's industry standards.

The SRS electric power systems must provide power on a reliable basis to accomplish the following safety functions:

- o Operation of reactor shutdown systems including control rods, safety rods, and supplementary safety system.
- o Circulation of D₂O coolant with adequate distribution to cool irradiated components in the reactor and operation of emergency cooling system.
- o Circulation of H₂O through the main heat exchangers.
- o Operation of airborne activity confinement system.
- o Discharge of irradiated components from the reactor, both during normal plant operations and for accident mitigation.
- o Monitoring of important plant variables.

DOE commissioned an independent safety assessment (ISA) of the electric power systems for the Savannah River production reactors. This report (Reference 1) recommended that a graduated-risk-acceptance approach be used to determine the acceptability of the electric power system for restart. A key point in this approach is that at one end of the risk spectrum there needs to be a very strong confidence that the systems can accommodate the postulated transients and accidents that could occur in the near future. At the other end of the risk spectrum, postulated accidents that are rare (that is, extremely low probabilities) do not necessarily have to be resolved completely prior to restart. This approach is consistent with that used by the NRC when considering the readiness of a licensed nuclear power plant to restart. The general restart criteria below are based upon this graduated-risk-acceptance approach.

The ISA provided a set of technical recommendations for improvements, which have been adopted by DOE for implementation by the SRS operating contractor. The satisfactory implementation of the technical recommendations from the independent safety assessment is generally expected to provide reasonable assurance that the graduated-risk-acceptance criteria will be satisfied and the plant will be ready for restart. The specific restart criteria below are provided to summarize the major actions that are necessary to assure that the general criteria will be satisfied.

The electrical power systems at a nuclear reactor facility do not provide a direct safety function. Rather the electrical systems provide essential power, which enables other plant systems to provide the safety functions. The electrical power systems are essential support systems; however, the electrical power systems provide support to so many direct safety systems that the electric power systems are critically important to safety. At licensed nuclear power generating stations, the electric power systems consist of an "offsite" power system and an "onsite" power system. At SRS, the corresponding systems are the "normal" power system and the "backup" power system.

Actual operating experience at SRS shows that the normal power system has failed to provide electric power to a reactor area at the rate of 0.7 per year (Reference 2). Since this probability is large, postulated accidents must include consideration of loss of normal power. Actual operating experience shows that the emergency diesel generators (EDGs), which provide emergency power to the 105 reactor buildings, have experienced a number of failures during actual demands. Based upon this actual experience, the best available estimate that both EDGs, as components, will fail is about 0.02 per demand (Reference 1). Since this probability is large, loss of all ac power to the emergency buses in the 105 building (i.e., the "mini-Blackout") must be evaluated.

The normal power system consists of connections to the commercial electric power grid, government-owned power producing facilities located on the Savannah River Site, and associated power distribution systems to the reactor area and within the area. The backup power system consists of a set of distributed emergency diesel generators, which provide power directly to specific equipment; a pair of centralized EDGs, which provide power to various safety loads in the 105 reactor building; a set of distributed batteries, which provide power directly to specific equipment; a pair of centralized batteries, which provide power to various loads in the 105 reactor building plus ac and dc power distribution systems.

The electric power systems for the production reactors at Savannah River include several special features such as automatic shedding upon degraded frequency, a special Containment Substation, and extensive use of automatic transfer switches which provide alternate power sources for important loads.

The ISA is the primary source document for the development of both the general restart criteria and the specific restart acceptance criteria. Therefore, by design, these criteria encompass the recommendations of the Independent Safety Assessment.

6.3.1.2 Restart Criteria

General Restart Criteria

- (1) A normal electric power system and a backup electric power system shall be provided to support the functioning of structures, systems, and components important to safety. The safety function of each electric power system (assuming the other system is not functioning) is to provide sufficient capacity and capability to assure that: (1) the reactor can be shutdown, (2) maintain necessary cooling, confinement, and other safety functions, and (3) provide essential monitoring.

The Normal power system shall consist of the combination of electric power purchased from the commercial grid and electric power generated by online capabilities on the Savannah River Site (specifically the 400-D power house), plus an AC power distribution system. [The former dedicated reactor area power generating capabilities (184-K, -L, -P) have now been shutdown.]

The Backup power system shall consist of a distributed set of standby emergency diesel generators (EDGs) which support specific equipment important to safety plus one pair of general standby EDGs for the 105 reactor building, a set of dc power sources, and ac and dc power distribution systems.

- (2) For normal operations (Condition I) and moderate frequency events (Condition II -- events out to $10E-2$ per year), the electric power system shall be capable of supporting the appropriate safety systems, even if the postulated worst-case single failure were to occur. For example, this category of events might include the "mini-Blackout" which involves the loss of normal power to the reactor area and the loss of both GM EDGs which provide backup power to the emergency buses in the 105 Reactor Building, but without any concurrent accident initiator.
- (3) For Infrequent events (Condition III -- events between $10E-2$ and $10E-4$ per year), the electric power system shall be capable of supporting the appropriate safety systems, presuming no failures in the electric power system. For example, this category of events might include a "small" LOCA (up to 1000 gpm break flow) with loss of Normal power.

For any event in this category that the electric power system appears to not be capable of supporting the safety systems, a Justification for Continued Operations (JCO) will be prepared. For any event in this category wherein the operating contractor can not develop an adequate JCO, alternate design concepts, procedural enhancements, training and other risk-reducing actions will be proposed for adoption prior to Restart.

DOE will evaluate each JCO, and alternate risk-reducing action, on a case-by-case basis.

- (4) For Limiting Faults (Condition IV -- events between 10^{-4} and 10^{-6} per year), the electric power system will be analyzed to determine the extent to which design basis events can be accommodated within the existing design capability. For example, this category of events might include the "station blackout" which involves the loss of all Normal AC power and loss of all standby Backup AC power EDGs. Another example would be the worst-case double-ended guillotine break LOCA with Loss of Normal Power.
- (5) Emergency procedures shall be prepared to address electrical power systems abnormal and emergency scenarios.
- (6) Periodic surveillance, testing, and maintenance procedures shall be prepared for the electrical power systems.

Specific Restart Criteria

(1) Analyses

- (a) The performance capabilities of the electric power systems shall be established and evaluated for applicable design bases events, which include:
 - o A complete loss of all Normal ac power to the reactor area without a concurrent accident.
 - o The "mini-Blackout" which involves loss of Normal ac power to the reactor area, including the four emergency buses in the 105 reactor building, coupled with loss of Backup power to these buses from the GM EDGs. None of the other Backup power sources is affected in this scenario.
 - o A small break in the process water system (LOCA) equivalent to a 1000 gpm break flow with loss of Normal ac power to the reactor area.
 - o A small break in the secondary cooling water systems (cooling water or service water) with the loss of Normal ac power to the reactor area.
 - o Loss of Normal ac power to the reactor area during Discharge operations.
 - o The "Station Blackout" which involves SRS-wide loss of all ac power from all sources, as they affect reactor operations. This means loss of ac power from the commercial electric grid and from the SRS power generating facilities, plus loss of ac power from all emergency diesel generators (such as the GM EDGs for the 105 building, and EDGs in standby service to various pieces of equipment such as the confinement system exhaust fans and the CW basin recirculation system).

- o A major break in the primary process water system with loss of Normal ac power to the reactor area.
- o A major break in the secondary cooling water system with loss of Normal ac power to the reactor area.

The analyses shall establish and evaluate the extent to which the electric power systems can adequately support the safety systems if the "single failure" were to occur. Single failure analysis is not required for Condition IV design basis events. In consonance with the principle of "Defense in Depth," the ultimate goal is to show that, even when degraded by the single failure, sufficient capability remains to provide all safety functions.

- (b) Load analyses for the GM EDGs shall be provided to establish the extent to which the electric power system can support the safety systems provided to accomplish the safety functions. These analyses shall include the identification and evaluation of: precisely which loads are important to safety, all momentary and intermittent loads, and all unique (non-redundant) loads.
- (c) Analyses shall be provided to establish the capabilities of the dc power system to support the safety functions, including during various "blackout" scenarios. These analyses shall identify which batteries are considered important to safety, evaluate the battery sizing calculations, and evaluate the adequacy of dc system components in systems that have been provided with new larger-capacity batteries.
- (d) Analyses shall be provided to demonstrate that the electrical protective devices for the loads and sources that are associated with automatic transfer switches have been adequately coordinated.

(2) Testing

- (a) A performance test shall be performed to demonstrate in an overall integrated manner that the Backup electric power systems can accommodate a complete loss of all Normal ac power to the reactor area, without any concurrent plant transient or accident. This test shall demonstrate the capability of the GM EDGs to accept the initial $[t_0]$ automatic loads for the non-accident case.
- (b) The maximum "second-step" starting capability of the GM EDGs shall be demonstrated by test. This test serve as an indicator of the maximum load that may be manually loaded onto the EDGs subsequent to the initial loads.
- (c) Testing shall be performed to demonstrate through repeated successful operations that "Single Unit" operation of the GM EDGs (that is, using one EDG and the EDG tie breakers to energize loads on all four emergency buses) is a viable and reliable option.

- (d) All batteries determined to be important to safety shall be maintained and tested periodically (including once prior to restart) in accordance with the requirements and recommendations of IEEE standard 450, or its equivalent for non-lead-acid batteries. Certain batteries shall be given the Acceptance/Performance Test (full discharge) in accordance with this standard prior to restart. To the extent justified by the operating contractor and approved by DOE, some batteries may be given only an appropriately conservative Service Test in accordance with this standard prior to restart, with full compliance to be achieved in the short term following restart.
- (e) The frequency of periodic test starting of the GM EDGs shall be reduced until load-testing capability is provided.

(3) Procedures

- (a) The adequacy of the interrelationships among the numerous emergency procedures related to loss-of-power situations shall be established.
- (b) The adequacy of the technical content of procedures related to loss-of-power situations and consistency among the procedures shall be established.
- (c) Appropriate precautions and guidance shall be developed and included in procedures related to manual loadings on the EDGs to assure that the EDGs will not be inadvertently overloaded and to provide the corrective actions to be taken in overload situations.
- (d) Central control room operators and 108 Building operators shall be adequately trained in the procedure changes resulting from the criteria above prior to restart.

6.3.1.3 Evaluation

The DOE staff conducted detailed discussions with WSRC, detailed walkdowns of the K-Reactor, and the review of numerous technical documents. This review encompasses the most recently issued Facility Safety Analysis Report (FSAR) in addition to technical reports, test procedures and results, operating procedures, and other documents provided by the operating contractor.

In an August 30, 1990 letter to DOE (Reference 3), WSRC listed those specific actions the Electrical Implementation Task Force determined to be needed to implement each of the recommendations from the ISA of the electric power system (References 1 and 4). On September 4, 1990, the WSRC Change Control Board approved Schedule Change Notice 0322 (SCN-0322), which defines the approved scope, schedule, and deliverables for the implementation of the recommendations from the ISA of the electric power system. The approval of this SCN paves the way for entry of these actions into the ROMP system. The staff views ROMP SCN-0322 as an overall summary of the implementation actions,

and the August 30, 1990, action items list as the more detailed identification of the planned actions. These two documents are key to determining whether the actions are expected to lead to the desired end results.

General Restart Criteria

In order to be able to address the General Restart Criteria, many of the activities required to achieve compliance with the Specific Restart Criteria need to be acceptably completed. Pending submittal from WSRC of the information needed to adequately address the General Restart Criteria, compliance with the General Restart Criteria is an open item.

Specific Restart Criteria

Analyses

Specific Restart Criterion (1)(a)

WSRC has committed to evaluating the six additional accidents specified in this restart criterion. The general guidance that WSRC is using to evaluate the accidents includes development of a detailed justification for the accident scenarios to be reviewed, why other accident combinations will not be evaluated, a determination of the expected frequency of the new scenarios, and issuance of an RTR (or other document) that documents the evaluation. The staff understands that the evaluation of the new design basis events will be documented prior to restart in formal reports, and that the actual revision of SAR Chapter 15 will be performed after restart. The action items list is weak for ISA recommendation 7.4.7 in that it does not include an action item to assess the adequacy of the plant response to these events and to determine if any improvements are needed. This must be acceptably addressed in order to satisfy this criterion.

Until WSRC officially transmits adequate documentation to DOE, compliance with this specific restart criterion is an open item.

Specific Restart Criterion (1)(b)

In Reference 3, WSRC commits to revising an existing load study to (1) identify all essential and non-essential loads on the emergency buses which the diesel will pick-up in a loss-of-coolant-accident and Loss of Normal Power, and (2) identify and quantify momentary loads and provide justification for any loads that are considered negligible. Also, a commitment has been made to include the new load lists in the revised emergency operating procedures for restart. In addition, WSRC has committed to performing a new load study to define the extent to which the GM diesels are capable of accommodating accident scenarios ranging from expected events to worse case scenarios. A new load study will be issued that evaluates the capability of the diesels to handle selected accident scenarios less serious than the limiting worst case LOCA. The report should document any assumptions made and also incorporate single failure criteria where appropriate. As part of the performance of this load study, interfaces will have to be established between the analysis and engineering groups to determine the accidents to be

considered. Also, a determination will have to be made of the loads required to mitigate the accident and the electrical system capability to power those loads.

Compliance with this restart criterion is an open item until the analyses have officially been transmitted to DOE and are determined to be acceptable.

Specific Restart Criterion (1)(c)

WSRC has committed to making a determination of the batteries important to safety and performing a review of existing battery calculations to assure that they establish an appropriate design basis for restart. If need be, those existing calculations should be revised. In situations where battery size would impact accident mitigation, emergency operating procedures will have to be revised. In addition, new calculations may have to be performed. The action items list does not include a commitment to consider blackout scenarios as part of the rereview of these calculations. The intent of the ISA recommendation was that a rereview be performed in light of the newly established blackout scenarios to determine if the capabilities of the various batteries are adequate. To accomplish this intent, the rereview of the battery calculations needs to be coordinated with the evaluation of the blackout scenarios. The battery calculations need to show that the batteries provide an adequate coping time for these events. WSRC has committed to analyzing the adequacy of electrical components in DC power systems that have (or will have) new batteries. A report should be issued which evaluates plant hardware such as cabling, circuit protection, etc., to ensure that DC power system components are adequate to accommodate new and larger batteries.

Compliance with this restart criterion is an open item until the required documentation has been officially transmitted to DOE and determined to be acceptable.

Specific Restart Criterion (1)(d)

WSRC has committed to performing a protective device coordination study. Based on reviews of the draft coordination study, the scope to which they worked to is less than the intent of the ISA recommendation. The ISA described in detail the fault transfer nature of ATSS and used this concern to precipitate a device coordination study in the vicinity of the ATSS. The intent of the recommendation was to perform a complete coordination study and institute fixes where bad coordination existed. WSRC has taken the study only to the extent of the fault transfer concern and has not considered a complete coordination to the extent of all power sources and all loads in the vicinity of the ATSS.

Compliance with this restart criterion is an open item until WSRC provides the staff with an acceptable study.

Testing

Specific Restart Criterion (2)(a)

The Startup Test Review Committee is responsible for establishing the requirements and acceptance criteria of the test in the form of a test outline. The Joint Startup Test Group is responsible for developing the procedure and documenting and reviewing the results of the test. The test outline and procedure are still under development.

Pending completion of the required testing, compliance with this restart criterion is an open item.

Specific Restart Criterion (2)(b)

Part of the test required to satisfy specific restart criterion (2)(a) involves measuring the initial load that comes up on the diesels. This test should take that initial load and add other loads in incremental succession to get an indicator of how much the diesel can stand in terms of the big inductive loads brought onto it in the event of an accident. The test outlines and procedures are still being developed for this test.

Pending completion of the required testing, compliance with this restart criterion is an open item.

Specific Restart Criterion (2)(c)

This test will be performed using each diesel several times to power a certain number of loads tied to the emergency buses. Each time both diesel will be running, and then one will be tripped with the tie breakers being used to connect its loads to the diesel still running. The test outline and procedures are still being developed for this test.

Pending completion of the required testing, compliance with this restart criterion is an open item.

Specific Restart Criterion (2)(d)

Through discussions between WSRC and DOE, decisions have been made on the list of safety-related batteries as well as which components of IEEE-450 will be applied before restart. Based on the types of some of the batteries and their use to provide a short duty cycle (on the order of seconds) some batteries will only be tested functionally by a short discharge. Other batteries will undergo a service or discharge test depending on which type may envelope the other or how new the battery is. At this time, DOE is in the process of reviewing the results of the tests. Based on the results of some of the tests, discussions are now taking place among the DOE test engineers and system engineers on how well these tests meet the intent of IEEE-450.

Based on the above, compliance with this restart criterion remains an open item.

Specific Restart Criterion (2)(e)

In K-Area, the diesels cannot easily be loaded for periodic testing because there is no capability to synchronize the generator frequency to normal power. Consequently, the periodic bimonthly startups of the diesels are performed with the diesel unloaded. WSRC has not changed the test frequency. Due to the problems of GM diesel #1 occurring from September 1990, to the present (March 1991) this issue will have to be revisited.

Based on the above, compliance with this restart criterion remains an open item.

Procedures

Specific Restart Criterion (3)(a)

DOE has reviewed both the Electrical Implementation Task Force action items list and the SCN definition of the approved scope and deliverables for procedures. Recommendation 7.2.6.1 requires the establishment of the proper interrelationships between the various emergency procedures related to loss of power situations. This includes assuring that appropriate linkages are provided and that the technical content of the procedures is appropriate, accurate, and consistent. The action items list indicates that the Implementation Task Force is planning to use flow charting to implement this recommendation, that appropriate revisions to the individual procedures will be identified, and that a final flow chart will be developed to depict the proper overall relationship among the loss of power procedures.

Pending the completion and issuance of these procedures, compliance with this restart criterion is an open item.

Specific Restart Criterion (3)(b)

WSRC is in the process of evaluating the applicable procedures. Pending the completion of the evaluation and the revision or development of necessary procedures, compliance with this restart criterion is an open item.

Specific Restart Criterion (3)(c)

WSRC has committed to developing a procedure for manually loading the GM EDGs that provides lists of essential loads and non-essential loads with the corresponding load rating in both amperes and kilowatts. Instructions for manually applying and removing these loads and appropriate precautions to assure that the GM EDGs are not inadvertently overloaded are also to be provided. In addition, existing procedures will be revised to reference this new procedure wherever manual loading is involved.

Pending the development and issuance of this procedure, compliance with this restart criterion is an open item.

Specific Restart Criterion (3)(d)

SCN-0322 Deliverable #6 indicates that operations personnel (both central control room operators and 108 building operators) will be trained on these procedure changes.

This restart criterion is acceptably satisfied.

6.3.4 Conclusion

As discussed above, there are many open items related to the specific restart criteria. Further, the general restart criteria cannot be addressed until after the satisfactory completion of many of the specific restart criteria. Resolution of the open items will be documented in a supplement to the SER.

6.3.1.5 References

1. J. T. Beard, Inc., Engineering Consultant, "Independent Safety Assessment: Electric Power Systems, Savannah River Production Reactors," May 21, 1990.
2. DuPont, Technical report DPST-85-984: J. A. Smith "Analysis of Principal SRP Electric Power Systems," dated October 31, 1985.
3. WSRC, Letter OPS-RRD-900093, "Implementation of Electrical Assessment Recommendations," dated August 30, 1990.
4. U. S. Department of Energy, Letter, Frank R. McCoy III, SRSP0 to John Gallagher, Westinghouse Savannah River Company, "Independent Safety Assessment of Savannah River Production Reactor Electric Power Systems," dated June 4, 1990.

6.3.2 Environmental Qualification

6.3.2.1 Background

Some design basis accidents may result in exposing equipment to harsh environmental conditions such as temperature, humidity, pressure, water spray, and radiation levels, that are significantly more severe than the environment that occurs during normal plant operation or during anticipated operational occurrences. Certain electrical equipment important to safety is relied upon to perform its safety function(s) during and/or after exposure to such harsh environmental conditions in order to prevent or mitigate consequences of the accident or to monitor post-accident conditions. Such equipment must be demonstrated to be capable of performing its required safety function(s) during and/or after exposure to those harsh environmental conditions.

DOE Order 5480.4, "Environmental Protection, Safety, and Health Protection Standards" (Reference 1), requires compliance with IEEE 308-1980, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations" (Reference 2) and IEEE 603-1980, "Standard Criteria for Safety Systems for Nuclear Generating Stations" (Reference 3). Both standards require equipment qualification to be in accordance with the requirements of IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" (Reference 4). The procedures described by IEEE 323-1974 are acceptable to the U. S. Nuclear Regulatory Commission (NRC) staff for satisfying the Commission's regulations pertaining to the qualification of electrical equipment for service in nuclear power plants to ensure that the equipment can perform its safety functions subject to Section 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," of Title 10 of the Code of Federal Regulations (CFR), Part 50 (Reference 5). In the commercial nuclear industry, the NRC recognized the importance of environmental qualification (EQ) to safety and required that specific EQ requirements be satisfied for all nuclear power plants. EQ was one of the largest retrofits ever imposed on the commercial nuclear industry, and compliance with the NRC's EQ requirements took several years to implement.

There are significant differences between commercial light water nuclear plants and the SRS reactor facilities that will result in substantially less harsh post-accident environments in the SRS facilities. However, the basic concept that equipment must be capable of performing safety functions under the environmental conditions in which it is relied upon to function remains valid and applicable to the restart of the K-Reactor facility.

6.3.2.2 Restart Criteria

The criteria for determining the adequacy of the SRS EQ program for restart are described below. This is a long-term program that is to be implemented after restart. However, the program and its schedule for implementation shall be provided to and approved by the DOE staff prior to restart. In addition, a risk-based justification for continued operation (JCO) shall be provided.

- (1) The long-term EQ program shall include the following.
 - (a) All electrical equipment within the scope of 10 CFR 50.49(b), as applicable, will be identified.
 - (b) The environmental conditions that could result from all postulated accidents identified in Chapter 15 of the Safety Analysis Report (SAR) will be defined. All environmental conditions that are harsh, as defined in 10 CFR 50.49, will be identified.
 - (c) All electrical equipment identified above will be categorized into one of the following categories:
 - i. Equipment required to function during and/or after an accident that results in its exposure to harsh environmental conditions.
 - ii. Equipment not required to function during and/or after an accident that results in its exposure to harsh environmental conditions, but where failure may be detrimental to plant safety or accident mitigation, or may mislead the operator.
 - iii. Equipment not required to function during and/or after an accident that results in its exposure to harsh environmental conditions, and whose failure is determined not detrimental to plant safety or accident mitigation, and will not mislead the operator.
 - iv. Equipment that performs its function prior to exposure to post-accident harsh environmental conditions and whose subsequent failure is determined not detrimental to plant safety or accident mitigation, and will not mislead the operator.
- (2) All equipment in category i will be documented to be environmentally qualified in accordance with the requirements of 10 CFR 50.49, and the restart criteria contained and referenced in the U. S. Nuclear Regulatory Commission Regulatory Guide (RG) 1.89 (Reference 6) and Section 3.11 of the U. S. Nuclear Regulatory Commission Standard Review Plan (NUREG-0800) (Reference 7).
- (3) As allowed by 10 CFR 50.49(k), all existing equipment will be qualified in accordance with the Division of Operating Reactors (DOR) Guidelines. More extensive use of analysis and operating experience is allowed for existing equipment. New equipment will be qualified in accordance with 10CFR50.49 requirements. For equipment not documented as qualified, a justification for interim operation (JIO) will be documented using 10 CFR 50.49(i), or the equipment must be replaced with suitably qualified equipment. Replacement equipment will be qualified in accordance with 10 CFR 50.49, unless there are sound reasons to the contrary, as identified in RG 1.89 and NUREG-0588, Category 1 (Reference 8).

- (4) Equipment in category ii will be documented to be qualified to the extent necessary to demonstrate that it will not fail in a manner detrimental to plant safety or accident mitigation, and will not mislead the operator. Equipment in categories iii or iv need not be environmentally qualified.
- (5) The mechanical equipment within the scope of 10 CFR 50, Appendix A, General Design Criteria 4, "Environmental and Dynamic Effects Design Bases" (Reference 9) shall be identified and qualified in accordance with Section 3.11 of the NRC Standard Review Plan (NUREG-0800), "Environmental Qualification of Mechanical and Electrical Equipment." The mechanical equipment shall be categorized, documented and qualified in the same manner as described above. Particular emphasis shall be placed on the applications of highly susceptible components in radiation environments, such as the use of teflon or teflon tape in mechanical equipment. Applications of these highly susceptible components shall be identified and evaluated for acceptability.

6.3.2.3 Evaluation

The Long Term Environmental Qualification Program (LTEQP), Revision 0, has been submitted to and reviewed by DOE. The program has been approved with comments which are to be implemented after restart. WSRC must provide a schedule for implementation of the comments and the LTEQP prior to restart. Pending receipt of the schedule of the LTEQP, this is an open item.

WSRC has developed a procedure entitled, "Guidelines for Preparation of Restart Environmental Qualifications Justification", to provide JCO's for equipment not qualified prior to restart. The individual JCO's prepared using this guidelines have been reviewed by DOE and found inadequate. WSRC has agreed to provide new JCO's incorporating all relevant DOE comments. Pending resolution of comments and DOE approval, the JCO program is considered an open item.

6.3.2.4 Conclusion

Based on the above evaluation, the DOE staff concludes that WSRC is in the process of developing a long-term EQ program to address the restart criteria. The staff will review the program schedule and its implementation when it has been completed and made available for review. The staff will also review the individual JCO's as they become available.

6.3.2.5 References

1. U. S. Department of Energy, DOE Order 5480.4, "Environmental Protection, Safety, and Health Protection Standards."
2. Institute of Electrical and Electronic Engineers, IEEE 308-1980, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."
3. _____, IEEE 603-1980, "Standard Criteria for Safety Systems for

3. _____, IEEE 603-1980, "Standard Criteria for Safety Systems for Nuclear Generating Stations."
4. _____, IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."
5. Title 10 of the Code of Federal Regulations, Part 50, Section 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."
6. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants."
7. _____, NUREG-0800, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment."
8. _____, NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."
9. 10 CFR 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."

6.4 Airborne Activity Confinement System (AACS)

6.4.1 Background

The containment system for commercial nuclear reactors is designed to provide a nearly leak-tight barrier against the uncontrolled release of radioactivity to the environment, even under pressure and temperature conditions of a postulated accident. In this design, the release of fission product and radioactive materials to the general public is prevented by the multiple-barrier concept of containment. At commercial plants, fuel integrity constitutes the first barrier against the release of radioactive materials; the reactor coolant system pressure boundary constitutes the second barrier; and the containment system functions as the final barrier.

The Savannah River Site (SRS) reactors are designed with an airborne activity confinement system (AACS) rather than a containment system. The AACS is an active ventilation system that continuously exhausts air from prescribed zones in the reactor building through a set of filters in order to prevent unattenuated release of contaminated air to the environment under accident conditions. The system is designed such that air flows from areas of least expected radioactive contamination to areas of higher potential for radioactive contamination. This is achieved by maintaining the process areas at negative pressures by the operation of fans and dampers.

The confinement structure is neither leak-tight nor designed to confine all gaseous radionuclides. Radiohalogens and particulates should be retained by the filters, but noble gases and tritium would be released during an accident. The release of these isotopes from design basis accidents must not constitute a significant risk to the public.

Due to the design differences between a confinement system and a containment system, a specific safety evaluation of the confinement system prior to K Reactor restart was deemed to be appropriate.

6.4.2 Restart Criteria

The criteria for determining the adequacy of the AACS at SRS for restart are described below.

(1) Environmental Conditions

- (a) The following environmental conditions shall be determined for the AACS during postulated design basis accidents (DBAs):
 - o maximum differential pressure from the anticipated pressure surge
 - o relative humidity of influent to the charcoal absorbers and filter elements
 - o maximum temperature of influent to the charcoal absorbers and filter elements

- o fission product source terms released to the confinement from postulated DBAs, including elemental and organic iodines and particulates (e.g., cesium, strontium, etc)
 - o integrated radiation dose for system components in the vicinity of the filters over a 30-day period following a DBA
- (b) The lowest and highest anticipated temperatures to which the AACS will be subjected shall be established.

(2) System Design Criteria

- (a) The reliability of the ducts, dampers, fans, instrumentation and filters shall be demonstrated.
- (b) A determination shall be made regarding the vulnerability of the AACS to single failures. Any identified failure scenarios shall be prioritized as to their likelihood of occurrence. An assessment shall delineate the degree of system performance degradation associated with each failure and the recommended compensatory action.
- (c) The AACS system and components shall function under the worst case environmental conditions defined under Restart Criterion (1).
- (d) The AACS shall be instrumented to signal, alarm and/or record various parameters associated with proper operation (e.g., flows, pressures, temperatures, radiation levels, fan and damper configurations).
- (e) Filter compartment water drains shall be demonstrated to be capable of meeting their design function.
- (f) The possible iodine desorption and autoignition that may result from radioactivity induced heat in the carbon beds shall be considered when determining the adequacy of the charcoal absorbers.
- (g) The absorber section of the AACS shall contain impregnated activated carbon demonstrated to remove gaseous iodine from influent. The carbon filters shall have an average atmosphere residence time of 0.25 seconds per 2 inches of absorbent bed. The maximum loading capability shall be 2.5 mg of total iodine per gram of activated carbon. No more than 50 mg of impregnate per gram of carbon should be used. The radiation stability of the type of carbon used shall have been demonstrated and certified.
- (h) Prior to restart, a plan shall be submitted for DOE approval to seismically qualify the AACS. System boundaries and equipment lists shall be generated to assure the comprehensiveness of the

plan. The plan should include a recommendation and its basis for either modification or replacement of the current filter compartments.

(3) Component Testing Criteria

Testing prior to restart is to be done in accordance with the current procedures. Before restart, WSRC shall submit for DOE approval a schedule which will result in compliance with the component testing criteria listed below after restart but before the next scheduled outage.

- (a) Demisters shall be tested by methods comparable to the requirements of American Nuclear Standards Institute (ANSI) Standard N-509 (Reference 1).
- (b) HEPA filters shall be dioctylphthalate (DOP) tested in accordance with provisions comparable to those specified in MIL-F-51068 and MIL-STD-282 initially and per ANSI N510 (Reference 2) in accordance with Technical Specification required frequencies. DOP penetration shall be less than 0.05 percent at rated flow.
- (c) Original and replacement batch material samples of the impregnated activated charcoal absorber beds shall be tested using methods comparable to American Society of Testing and Materials (ASTM) Standard D3803, Method D. Additionally, test samples shall be withdrawn from the filter beds and tested in accordance with Technical Specification requirements.
- (d) The activated carbon section of the AACS filters shall be leak-tested with halogenated hydrocarbon refrigerant using methods comparable to those specified in ANSI N-510 to ensure bypass leakage through the absorber section is less than 0.05 percent. The test shall be conducted at installation of new filter elements in accordance with Technical Specification frequency requirements.
- (e) The air flow distribution through the AACS filter units shall be tested for uniform flow distribution per ANSI N510.
- (f) The AACS filter compartments shall be leak-tested periodically per the provision comparable with ANSI N-510 specifications.
- (g) All active components of the AACS shall be tested periodically and the results documented. Frequency of testing shall be at least as often as specified in Technical Specifications.

(4) Maintenance

- (a) A preventive maintenance program shall be specified for both active and passive components of the AACS. The program shall include dampers, fans, power supplies, air supplies, instrumentation, and controls.

- (b) The use of silicone sealants or any other temporary patching materials on filter compartments shall not be allowed.

6.4.3 Evaluation

Environmental Conditions

The environmental conditions expected during and after design basis accidents have not been provided. This is an open item.

System Design Criteria

Restart Test Identification Report (RTIR) No. 10 (Reference 3) identifies the safety functions of the AACS and the required restart testing that will demonstrate equipment operability. A direct linkage of safety function to test requirement is provided in the RTIR to ensure all safety functions are tested. Specific safety functions to be tested include the following:

- o process-room negative relative pressure with total air flow at least 40,000 cfm
- o ignition temperature of in-service carbon filter material at least 250 degrees C
- o 99 percent removal efficiency for 0.6 micron particles for high-efficiency particulate filters
- o system monitoring and controls
- o carbon removal efficiency at least 99.25 percent for elemental iodine and instantaneous bypass at most 0.5 percent
- o dampers fail conservatively on loss of power or control air
- o air flow distribution within filter compartments

Specific test procedures are identified in the RTIR that will fulfill the testing requirements. The staff considers the RTIR to adequately document the required equipment testing to demonstrate equipment operability. The completed test procedures will be reviewed upon completion of system startup testing to verify successful equipment testing. This is an open item.

The Report of the Special Defense Program Filter Review Group (Reference 4) documented a thorough review of system single failure vulnerabilities. Potential system vulnerabilities included the following:

- o Failure of an exhaust fan damper to close when its associated exhaust fan stops, resulting in possible back flow through the fan

- o Inadvertent opening of the -40' level to exhaust stack damper, resulting in unfiltered release of below-grade air to the environment and loss of negative relative pressure in process areas.
- o Inadvertent filter compartment supply damper closure following absorption of fission products on filter elements, resulting in loss of required cooling to filter elements.
- o Starting of supply fans could result in process area pressurization.
- o Spurious -40' level exhaust damper closure could block exhaust of below-grade process air through the stack, resulting in ground level releases.

The study recommended various procedural and hardware fixes to address these vulnerabilities. WSRC was requested to address these recommendations. The recommendations were considered, and a WSRC response was received. DOE did not concur with the WSRC response and the points of nonconcurrence were documented in a letter to WSRC. A new response to the recommendations from WSRC is necessary. Single failure vulnerability is an open item.

ROMP item WISR-35 requires replacement of bearing, grease, and fan belts for the ventilation fans with more temperature- and radiation-tolerant equipment. WISR-35 also requires replacement of fan motor insulation with a more temperature-tolerant winding and modification of the motor protection circuit to install more temperature-tolerant breaker trips. The staff cannot assess the adequacy of these upgrades until the system's environmental conditions during and after design basis accidents are provided. This is an open item.

Chapter 6.2.2.5 of the Safety Analysis Report (SAR) lists the Central Control Room instrumentation associated with the AACS. The instrumentation list in the SAR was compared by the staff against the system's safety functions as listed in the RTIR. No deficiencies were found. The staff considers the system instrumentation to be adequate to support reactor restart.

Restart criteria associated with filter compartment water drain capacity, charcoal absorber adequacy, and absorber design and radiation stability have not been addressed. These are open item.

DOE has committed to the Defense Nuclear Facilities Safety Board to substantially accelerate its program for seismically qualifying the AACS. Prior to restart, procurement of new, seismically rugged, filter compartments will be initiated and Seismic Qualification Utilities Group (SQUG) walkdowns will be initiated in K-Area to identify seismic bracing deficiencies. Design of modifications to fix SQUG-walkdown identified deficiencies will be initiated. Complete seismic qualification of the system is now expected by the second cycle outage after restart.

Component Testing Criteria

The AACS components will be tested prior to restart per the Restart Test Program. Testing requirements are now being identified, and applicable procedures are being written. WSRC needs to provide a schedule and plan for revised testing per industry practices as outlined in American National Standards Institute (ANSI) Standards ANSI N-510 and ANSI N-509. Adequacy of the post-restart testing criteria and schedule is an open item.

Maintenance

A draft of the preventive maintenance plan has been received. Until a final acceptable program is provided, adequacy of the preventative maintenance program is an open item.

The criterion related to the use of silicone sealants or any other temporary patching materials on filter compartments has not been adequately addressed by WSRC. This is an open item.

6.4.4 Conclusion

Based on the above evaluation, the DOE staff concludes that many of the activities and resulting documentation to address this issue have not been completed. Resolution of the open items identified above will be documented in a supplement to this SER.

6.4.5 References

1. American National Standards Institute, ANSI N509-1989, "Nuclear Power Plant Air-Cleaning Units and Components."
2. American National Standards Institute, ANSI N510-1980, "Testing of Nuclear Air-Cleaning Systems."
3. WSRC, "Airborne Activity Confinement System," Restart Test Identification Report No. 10, Revision 0, March 15, 1990.
4. Knuth, D. F., et al., "Review of the Savannah River Site Reactor Confinement System," February 28, 1990.

6.5 Safe Shutdown

6.5.1 Background

Reactors are designed for safe shutdown to ensure that the integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor and sustain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of design basis events (DBEs) or design basis accidents (DBAs) is maintained. Safe shutdown systems, including remote shutdown systems, support systems, and associated equipment must be capable of maintaining the reactors at a satisfactory margin below criticality; providing uninterrupted, long-term decay heat removal capability and monitoring; and maintaining safe shutdown during all design bases normal, abnormal, and accident conditions.

To ensure that commercial reactors maintain the capability to safely shut down, they are designed to meet U.S. Nuclear Regulatory Commission (NRC) General Design Criteria (GDC) in Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix A (Reference 1). The Savannah River Site (SRS) reactors, however, were designed and constructed before the GDC for commercial nuclear reactors were developed. Recently, the DOE specified per DOE Order 5480.6 (Reference 2) that the "General Design Criteria specified in 10 CFR 50, Appendix A shall be applied to all DOE-owned reactors" if "safety can be significantly improved by implementing one or more of the criteria." The SRS reactors are now in a state of transition, as the plants are upgraded to meet applicable GDC.

Safety margins for the SRS reactors can be enhanced by selectively applying the GDC. The following sections of this evaluation will describe the applicability of the GDC to SRS reactors to ensure the reactors are safe to operate. In addition, current programs, processes, and activities will be evaluated to determine if they satisfy the restart criteria.

To determine which plant systems are necessary to ensure safe shutdown of the SRS reactors, all DBEs and DBAs must be evaluated to determine the plant systems necessary to mitigate the effects of these events. The effects of single-failures (see 10 CFR 50, Appendix A, "Definitions") (Reference 3) on the safe shutdown capability must also be evaluated. Chapter 15 of the SRS Safety Analysis Report (SAR) lists the DBEs and DBAs that must be addressed to determine the capability for safe shutdown of the SRS reactors.

6.5.2 Restart Criteria

The criteria for determining the adequacy of the SRS safe shutdown for restart are described below.

- (1) Systems shall be provided to bring the reactor to a safe shutdown condition and to prevent high radiation doses from being released to the environment and public during normal operation or in the event of an accident.

- (2) WSRC shall ensure that safe shutdown systems are capable of:
- o maintaining the reactors at a satisfactory margin below criticality
 - o providing adequate long-term decay heat removal capability and monitoring
 - o maintaining these conditions during all postulated, design bases normal, abnormal, and emergency conditions
- (3) Local and remote safe shutdown systems shall be sufficient to ensure that no single failure (as defined in 10 CFR 50, Appendix A, "Definitions") could result in the loss of the ability of either of these systems to achieve and maintain safe shutdown. Shutdown systems installed to ensure postfire shutdown capability need not be designed to meet the single failure criteria or other design basis accident criteria, except where required for reasons other than fire (i.e., because of adverse systems interactions or spurious controls operations from fire or fire suppression system actuations). Safe shutdown systems shall be periodically tested to ensure operability.
- (4) During all credible events and accidents, as well as during normal operation, systems required for safe shutdown shall be able to function in the event of either a loss of offsite power or a loss of onsite power (onsite and offsite power as defined in Section 6.3.1, "Electrical Power and Control Systems"). Onsite power for essential safe shutdown systems shall be available for 72 hours following any design basis event or accident.
- (5) Should the control room become uninhabitable, there shall be controls and monitoring equipment provided that are physically remote from the control room for achieving shutdown and maintaining the reactor in a safe shutdown condition, including the ability to maintain uninterrupted, long-term cooling. There shall be remote controls and monitoring equipment consistent with the design requirements discussed above.
- (6) Access and egress to vital safe shutdown monitoring and control stations shall be available during all credible environments and events. Emergency lighting qualified to credible environments and events, and is available for at least 72 hours, shall be provided for these routes, as well as for the safe shutdown monitoring and control stations. The battery supply for these lights shall be qualified for at least 8 hours at the lowest allowable voltage level.
- (7) Communications between safe shutdown monitoring and control stations and appropriate emergency stations shall be available and capable of performing their intended functions during and after credible environments and events, including loss of offsite power. The

communications equipment shall be controlled such that it will not introduce unacceptable spurious systems actuation due to electromagnetic induced effects.

- (8) Identified potentials for spurious operations of equipment that could prevent the safe shutdown of the reactors shall be evaluated. On the basis of their significance, the potentials for spurious operation shall be eliminated before restart, or a justification for not eliminating them before restart shall be provided to DOE for review and approval.
- (9) WSRC shall complete the actions related to essential core monitoring as stated in the "Westinghouse Independent Safety Review of Savannah River Production Reactors" (WISR), Item 19.
- (10) There shall be means available to reliably shutdown the reactor fast enough to halt the fastest anticipated reactivity transient that could damage the reactor or assemblies.
- (11) The reactors shall be in compliance with all DOE Orders determined to be necessary by restart by the Order Compliance Joint Task Force (reference Section 2.5).
- (12) Safe shutdown systems shall be environmentally qualified in accordance with the criteria listed in Section 6.3.2.

6.5.3 Evaluation

The Reactor Operations Management Plan (ROMP) (Reference 4) states in Item SE-6.5, "Safe Shutdown," that a Technical Position Paper (TPP) will be provided to DOE to address the safe shutdown criteria. WSRC provided an outline for the TPP, which the staff reviewed. The staff determined that the outline addressed the acceptance criteria. The outline was incorporated into the ROMP, as Revision 4. The technical position paper is currently under internal review by WSRC, and there is no firm delivery date to DOE.

Completion of the internal review is not expected until the SAR and Technical Specification updates have been completed. A draft of the TPP, dated October 2, 1990, was provided to DOE and was reviewed by the staff. The Safe Shutdown TPP includes two major sections. These major sections are described in the following paragraphs.

The section, "Summary Description of the Safe Shutdown Systems," provides a description of the systems required to achieve a safe shutdown for each of the major situations within the design basis for the production reactors. It includes a background or introductory subsection describing the fundamental requirements to be met: subcriticality, decay heat removal, essential monitoring, essential lighting, habitability systems, and communications systems. A table is included depicting major situations, events, and accidents and the systems necessary to meet the safe shutdown fundamental

requirements. The accidents listed in the table bound other less severe incidents not included in the table. Events included in the table are: normal shutdown, reactivity insertion accident, loss of normal power (LONP), earthquake/LONP, loss of coolant accident (LOCA), loss of cooling water/loss of pumping accident, and fire/LONP. For each event included in the table, a separate subsection describes the functioning of the shutdown systems during the event. Appropriate references support the descriptions. Each event is assumed to extend over a 72-hour period.

A description of the systems needed to achieve safe shutdown if the control room should become uninhabitable is provided. Fuel Discharge accidents are not included because the reactor is already in a safe shutdown condition during such operations, and the SAR includes the melting of a dropped fuel assembly as a design basis accident. Spurious controls operation is addressed only for fire and earthquake events.

In support of the descriptions, a table listing disposition of relative restart issues is included. The table identifies the issue, its concern, its disposition, and its resolution, if required prior to restart. The table is structured around the fundamental requirements for safe shutdown.

The section, "Conformance With Restart Criteria," includes an explanation of how each safe shutdown criterion is met. Separate subsections address each criterion with references to section 2 and other documents as necessary. These subsections are not lengthy, because the descriptions in section 2 provide objective evidence (including references) that the various criteria are met. As appropriate, each subsection addresses each of the major events as it relates to the particular criterion covered by that subsection.

The staff's review of the draft safe shutdown TPP indicates that all of the restart criteria in the SER appear to be addressed. However, compliance with the restart criteria for this section can not be fully evaluated until the final Technical Position Paper on Safe Shutdown is received. This is an open item pending review of the final TPP.

6.5.4 Conclusion

Based on the evaluation of the draft TPP, the staff concludes that WSRC is developing a Technical Position Paper that will address all the restart criteria for safe shutdown. The level of compliance with the restart acceptance criteria cannot be judged at this point. The staff will review the programs developed to meet the restart criteria for safe shutdown reliability when they have been completed and are available for review.

6.5.5 References

1. Title 10 Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria."
2. The U. S. Department of Energy (DOE) Order 5480.6, "Safety of Department of Energy Owned Nuclear Reactors."

6.6 Sump Water Removal

6.6.1 Background

The purpose of the contaminated water removal and disposal system (CWRDS), i.e., the sump water removal system, is to remove and retain the emergency coolant or cooling water following a design basis loss of coolant accident (LOCA) or seismic event. This system operates in conjunction with the Airborne Activity Confinement System (AACS) to control the release of liquid and airborne contaminants.

The main sources of water for the CWRDS are process water (D_2O) leaks, cooling water (H_2O) leaks, emergency spray water, and emergency coolant from reactor pressure relief and overflow lines. The CWRDS consists of drains, vents, sumps, pumps, valves, and tanks dedicated to the removal and containment of water originating from any of the aforementioned contaminated water sources. The sump pumps play an integral role in ensuring that the DC motors for the process water pumps are not flooded out during a LOCA.

6.6.2 Restart Criteria

The criteria for determining the adequacy of the SRS containment sump removal system for restart are described below.

(1) Performance

- (a) The pump room sump pumps, which consist of two main nonsubmersible and two submersible sump pumps, must have sufficient capacity (in conjunction with emergency cooling system throttling) in design basis loss of coolant accidents to prevent flooding of process water DC motors until flow supplied by the DC pumps to the reactor core is no longer necessary to maintain cladding integrity.
- (b) The sump pumps shall have adequate protection from trash or foreign objects which may be present under flooding conditions.
- (c) Flood levels during events that might require the containment sump removal system to be functional shall be determined. For those portions of the containment sump removal pumps or their associated equipment (e.g., motors, electrical connections) that are required to operate while submerged, the equipment shall be demonstrated to function during the postulated submergence.

(2) Seismic

The sump water removal capability during and following a seismic event must exceed the largest leak rate from a single non-seismically qualified cooling water line per ANSI/ANS 58.2-1988.

(3) Testing

The sump pumps must have periodic tests which adequately indicate their actual discharge capacity in an emergency situation. Pump head/flow curves may be used in conjunction with reduced flow testing if full flow testing is not possible. The curves to be used must have been previously verified by testing at the full flow level or have been provided by the vendor for the specific model/type installed. Pump test procedures must identify test acceptance criteria (i.e. acceptable head/flow curves and conditions).

(4) Redundancy/Separation

The system must perform its design function with the failure of any single active component.

(5) Radiological Considerations

WSRC shall verify that operation of the sump water removal system during DBA conditions shall not result in whole body and thyroid dose exposure exceeding Title 10 of the Code of Federal Regulations (CFR), Part 100 siting criteria. The analysis should be based on 100 percent of the reactor heavy water inventory reaching the retention basin following the DBA.

6.6.3 Evaluation

Performance

A report on sump water removal capability was issued jointly by the Savannah River Laboratory (SRL) and Idaho National Engineering Laboratory (INEL), and Sandia National Laboratory. The report scope includes calculations on flow to the minus 40-ft level during credible break size LOCAs to prove the existing sump pump capacity will remove ECS flow to the minus 40-ft level and to show that loss of pumping accidents are tolerable for small break LOCAs. This report, issued September 25, 1990, identifies the worst case LOCA as a Process Water pump suction bellows failure in loop # 6 with four ECS injection paths (Reference 1). This report does not consider ECS throttling; therefore, this LOCA condition will cause an eventual flooding of the DC motors.

The combined design capacity of the sump pumps totals 14,000 gpm. This number is derived from Allis-Chalmers curve E-7188-3 for the main sump pumps, showing a rated capacity of 4,500 gpm at 95 feet TDH, and from Gorman-Rupp curve S8A1-1 for the submersible sump pumps, showing 2,500 gpm at 67 feet TDH. The acceptability of the 14,000 gpm flow capacity, with 5,000 gpm being submersible capacity, depends on expected leak rates and operator actions following ECS actuation.

The design basis accidents for the ECS are a loss of coolant accident from a double-ended guillotine break (DEGB) of a large process water line (D20) and a loss of pumping accident from a cooling water line break. Section 6.2.3.3.1.1 of the Safety Analysis Report (SAR) states that the ECS addition rate for a

3. Title 10 Code of Federal Regulations, Part 50. Appendix A, "Definitions."
4. WSRC, WSRC-RP-89-368, Reactor Operation Management Plan, Safety Evaluation 6.5, "Safe Shutdown."

LOCA is 16,400 gpm. The SAR does not quantify the ECS addition rate after a LOPA, but it should be comparable to the addition rate during a DEGB LOCA.

The emergency spray system (ESS) is manually activated per DPSOL 105-MC-2-LK upon ECS actuation. The ESS discharges spray water into the process room with a capacity of 2,100 gpm, per the SAR. Approximately 500 gpm is directed to the minus 40-ft pump room sumps through 6-inch standpipes in the process room. DPSOL 105-MC-2-LK directs ESS use until 30 minutes after nuclear shutdown. Therefore, about 16,900 gpm of coolant could be discharged to the pump room sumps with a LOCA occurring in the lower levels of the reactor building.

Recommendations to modify the ECS throttling procedures were to ensure operators are properly prioritizing the various operating data they are monitoring, such as assembly maximum temperatures, ECS flow rates, and sump levels. Reference 2 states that during a PW DEGB LOCA, the ECS flow can be throttled to 7000 gpm after approximately 1 hour, based on reactor core temperature limits. Per Reference 3, if ECS is throttled to 8000 gpm at or before the water level in the motor room reaches 30 inches, flooding of the DC motors will not occur.

DPSOL 105-MC-6, Revision 17 (Reference 4) is the emergency procedure used to throttle ECS flow. The procedure lists (Table 1 of Appendix 11) the minimum ECS flow rates for breaks in various locations of the process water system. The procedure allows ECS throttling only if certain conditions on process water flow, reactor shutdown time, and leak location isolation are met. In addition, at least one condition on cooling water basin level, ECS flow time, or sump level has to be met to allow ECS throttling. The justification for the most recent throttling limits has been submitted by WSRC and reviewed by the staff. The staff review of this procedure and its technical basis will be documented in Section 11.3 of this SER ("Power Limits").

ROMP Item SE-6.6 is the primary vehicle that addresses the restart criteria for this SER section. The ROMP committed to delivery of a technical position paper that will address the adequacy of the system's design based on expected leak rates. This position paper has not been issued for review. However, based on the above information, it appears that the Water Removal System will be shown to be adequate to prevent flooding of the DC motors during the worst possible PW LOCA event, if the decision and execution to throttle ECS flow is made at the appropriate time. This issue is an open item.

Design Request Memorandum 114, Revision 1 (Reference 5), lists the design requirements for debris screens to be installed around the sump pumps prior to reactor restart. The screens were installed around the pump room/motor room dams, the sump pumps, and the minus 30-ft balcony in the pump room. The screens are designed to allow each sump pump's maximum capacity even with the screens 75 percent blocked due to debris. This percentage was based on conservative, qualitative judgement. No credit is taken for horizontal surfaces on the screens. The screens are designed to be seismically rugged. The staff concludes that the design of the debris screens should adequately protect the sump pumps during flooding events.

Seismic

RTR-2650 (Reference 6) calculates the maximum light water leak rate to be expected from a single line after a design basis earthquake (DBE). RTR-2615 (Reference 7) calculates the expected process water leakage after a DBE. RTR-2637 (Reference 8) provides additional quantitative data in support of RTR-2650.

The ROMP requires that a seismically qualified moderator recovery system be operational prior to reactor restart. RTR-2615 indicates the maximum credible process water leak rate following a DBE is about 50 gpm from miscellaneous flanges and the pump seal system, since the Process Water System will be demonstrated to be seismically qualified. Section 9.3.7 of the SAR states that the design capacity of the Moderator Recovery System is 5 gpm to 1,000 gpm. Therefore, the sump pumps are not needed to control process water leaks following a seismic DBE.

RTR-2637 identifies a filtered water line as the largest nonseismically qualified cooling water line, with a flow rate of about 5,200 gpm. Industry experience is that piping is extremely resistant to failure from seismic loading so that the assumption of the failure of the largest nonseismically qualified cooling water line, with no other breaks, is adequate (Reference 9). RTR-2637 estimates the flow rate from miscellaneous flanged pipe joints to be 60 gpm. RTR-2637 further calculates that if the limiting filtered water line could be isolated within 1 hour of the DBE, an additional 29 hours would be required to flood the DC motors. For this particular filtered water line, a manual isolation valve that could be used to isolate the leak is located in the emergency pump room at the minus 20-ft level.

The DC-motor floodout calculations in RTR-2637 do not take credit for water removal via the sump pumps. The ROMP requires seismic qualification of the submersible sump pumps, their associated piping, and power and control circuitry prior to startup. The report, "Basic Data for Project S-4309," Revision 1 (Reference 10), requires seismic qualification of the level controls and power supplies for the submersible sump pumps. Modifications that allow the assumption of survivability for 5,000 gpm of submersible sump pump capacity following a DBE were installed during this project.

The isolation of the limiting filtered water line following a DBE within 1 hour is judged by the staff to be adequate. Since 420,000 gallons of water are needed to floodout the process water DC motors, the addition of 5,200 gpm from the filtered water line with concurrent removal of 5,000 gpm from the submersible sump pumps results in a flood out time of over 2,000 hours. Further, the assumption of the loss of the single largest nonseismically qualified cooling water line is justified based on industry experience. Therefore, the staff determined that qualification of the submersible sump pumps is sufficient to address miscellaneous leakage following a DBE.

Testing

The safety functions of the sump water removal system will be tested prior to restart per the Restart Test Program. Testing requirements are identified in

Restart Test Identification Report (RTIR) No. 7 (Reference 11). RTIR No. 7 identifies the safety functions of the CWRDS and the required restart testing that will demonstrate equipment operability. Safety functions to be tested include pump capacity, diversion valve function, and equipment control and monitoring functions. A direct linkage of safety function to test requirement is provided in the RTIR to ensure all safety functions are tested. Specific test procedures are identified in the RTIR, which will fulfill required testing. The staff considers the RTIR to adequately document the required equipment testing to demonstrate equipment operability.

The completed test procedures will be reviewed upon completion of system startup testing to verify successful equipment testing. This is an open item.

Redundancy/Separation

The ability to throttle ECS flow allows for single failure of a sump pump. Other single failures of active components related to operation of the sump pumps (e.g., sump level sensors, power supplies, and control circuits) are bounded by the failure of a single sump pump. However, small sump pumps are used in the sumps to discharge miscellaneous water accumulation during normal operation. The diversion valves downstream of the small sump pumps are normally positioned to allow this miscellaneous building water to be pumped to the process sewer after sampling. In a flooding accident, the diversion valves are diverted to the contaminated water storage tank by the Automatic Incident Action System. The single failure of a diversion valve would possibly result in several hundred gallons per minute of contaminated water being discharged directly to the process sewer. The environmental impacts of this possibility have not been assessed by the contractor. No impact on reactor safety would result from failure of the diversion valve.

Schedule Change Notice 229 of the ROMP anticipates the completion of a study on the necessity of diversion valve redundancy in November 1990. Capital equipment project S-1500, now funded, will replace the existing diversion valves with valves of an improved design. For restart, and prior to valve replacement, mitigating measures, such as operator action to disable the small sump pumps if the diversion valves have not diverted during incident action, should be in place. Alternatively, the risk of contaminated water being discharged to the process sewer via the diversion valves may be quantified and accepted, if proven to be low enough. The study and the corresponding IMC recommendation will be reviewed by the staff when they become available. This is an open item.

Radiological Considerations

This restart criteria will be addressed in the revision to Chapter 15 of the Safety Analysis Report. (See Section 11.1, "Transient and Accident Analysis" of this SER.)

6.6.4 Conclusion

Based on the above evaluation, the DOE staff concludes that WSRC is in the process of performing several activities that, when satisfactorily completed,

should adequately address the restart criteria. Resolution of the open items identified above will be documented in an SER supplement.

6.6.5 References

1. Idaho National Engineering Laboratory, EGG-EAST-9271, "Results of TRAC L-Reactor Bellows Break Calculations."
2. WSRC, WSRC-90-352, "Guidance for Long-Term Cooling for DEGB LOCA."
3. Horton, W. H., SRL-NES-900047, "Initial Work on LOCA Throttling and Flooding Implications," August 13, 1990.
4. WSRC, DPSOL 105-MC-6-PLK, "Operation of Reactor Cooling Facilities During an Incident," Rev. 17, January 12, 1990.
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6.7 System Integrity

6.7.1 Background

The K-, L-, and P-Reactors at SRS were designed, fabricated, and installed in the late 1940s to early 1950s. The reactor systems and components have been susceptible to in-service degradation during their long service period. Their susceptibility has been also influenced by the chemical composition, thermal-mechanical history of the steel, and the stresses acting on the steel. Moderator leakage, later attributed to intergranular stress corrosion cracking (IGSCC), was observed within 18 months of reactor startup. These early leaks were from cracks that developed in small-diameter piping. Subsequently, in addition to IGSCC, other small, through-wall cracks in the pressure boundary were attributed to other forms of corrosion, such as transgranular stress corrosion cracking (TGSCC), fatigue cracks, and wear. Recognizing the age of the SRS reactors and the evidence of material degradation, the Savannah River Laboratory (SRL) undertook an extensive evaluation known as the Reactor Materials Program (RMP) to better characterize the system condition and provide a comprehensive assessment of structural integrity of the SRS reactors.

The three reactors have been shut down since 1988. In order to restart the reactors, it has become necessary to evaluate the SRS reactor system integrity to provide a confirmation that the SRS reactors have the ability to maintain structural integrity under both normal and credible abnormal conditions. The following are the major topics relative to system integrity, and are the primary subjects of this evaluation:

- o influence of IGSCC present in the Process Water System (PWS)
- o WSRC analysis of the leak-before-break (LBB) capability and margins of safety of the various PWS components
- o in-service inspection (ISI) program for inspection and testing of the reactor systems and components
- o capability and sensitivity of the leak detection system

Overall structural integrity of the SRS reactors has been examined by the Operating Contractors (DuPont and WSRC) through the RMP, Topical Reports of Structural Integrity of the PWS, ISI programs, and various safety feature enhancements of the reactor systems. These modifications and upgrades are to ensure the adequacy of the margins of safety for the reactor operation and to remove the uncertainties caused by some material degradation in the principal safety systems resulting from the lengthy operation of the SRS reactors.

This section evaluates the structural integrity of the pressure boundaries of the PWS, the Engineered Safety Feature (ESF) Systems, and the Cooling Water System (CWS). Evaluation of the ESF system structural integrity includes the following safety systems:

- o Emergency Cooling System (ECS)
- o Moderator Recovery System (MRS)
- o Supplementary Safety System (SSS)

The restart criteria listed in Section 6.7.2 have been developed to address the applicable parameters including materials, engineering and design, fabrication process, water chemistry, inspection and testing, and leak detection.

6.7.2 Restart Criteria

The criteria for determining the adequacy of the SRS System Integrity Program for restart are described below.

(1) Process Water System (PWS)

WSRC shall meet the following criteria with respect to the system integrity of the primary coolant system.

(a) Engineering Analysis and Design

Demonstrate that the structural safety margins stated in the SAR are consistent with current plant configuration and will be maintained. These margins should be consistent with the national codes and standards pertaining to seismic (discussed in Section 6.1 of this report) and other postulated design basis accidents (discussed in Section 11.1 of this report).

(b) Fabrication and Modification Process Controls

Because austenitic stainless steels are used for construction of primary coolant boundary components in the Process Water System, process controls must be exercised during all stages of component manufacturing and system modifications to avoid severe sensitization of the material and to minimize exposure of the stainless steel to contaminants that could lead to stress corrosion cracking.

(c) Water Chemistry

Primary system water chemistry shall be controlled by preventing contamination and through purification to prevent IGSCC. Procedures shall be established to ensure that chemical parameters are controlled within the limits specified in the SAR. In addition, corrective actions shall be stated in case of chemistry excursions under normal operating conditions. Formal controls shall be established regarding the composition/purity of all reagents and make up chemicals introduced into the primary system. Procedures shall also be established for chemistry control and monitoring to prevent IGSCC under lay-up and shutdown conditions.

(d) In-Service Inspections (ISI) and In-Service Testing (IST)

The material and structural integrity of the primary system shall be demonstrated and maintained as defined in the SAR. An ISI program consistent with that discussed in the Inspection and Testing Section (SER Section 14) shall be implemented before K-Reactor restart. The program shall include visual, volumetric, and surface inspections as defined in ASME Section XI to the extent practical for implementation at SRS. All components and associated piping through the first isolation valve shall be inspected prior to restart.

Nondestructive examination techniques (ultrasonic testing (UT) and eddy current testing (ET)) shall be used to demonstrate that the reactor tank and the PWS piping will maintain their integrity under the operating conditions. The restart criteria for the reactor tank, based on the UT inspection, will be in accordance with WSRC-RP-89-208, "Tank UT Acceptance Criteria" (Reference 1).

Administrative controls and procedures shall be developed and implemented to provide guidance on repairs/replacements criteria on degraded piping and components identified through the ISI program.

(e) Leak Detection

Establish justifications for a sensitive leak detection capability to detect specified small leaks. Demonstrate that this leak detection capability is consistent with technical specifications that specify the operability of leak detection equipment. Conduct appropriate operator training on the leak detection system. Demonstrate that, for detectable leaking cracks, crack growth is slow enough to allow sufficient time to shutdown the plant. Also show that the through-wall cracks will leak and not plug as a result of impurities or corrosion, or ensure that appropriate margins are used to account for uncertainties.

(2) Engineered Safety Feature (ESF) System

Demonstrate that ESF systems have a low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The fluid and material compatibility for the ESF systems, and for the auxiliary systems that directly support the ESF systems, should be considered to reduce or eliminate the propensity of contaminant induced piping and component degradation. The ESF systems include the Supplemental Safety System (SSS), Moderator Recovery System (MRS), and the Emergency Cooling System (ECS).

(3) Cooling Water System (CWS)

Demonstrate that structural integrity of the CWS is maintained under normal and abnormal load conditions using applicable codes, standards, and analysis methods, including those specified in the seismic section of the SER. Components which are part of the CWS shall be subject to:

- (1) periodic inspection and testing of areas important to safety to

assess their leak tight integrity, and (2) periodic evaluation based on surveillance of the condition of the material.

6.7.3 Evaluation

During its evaluation, the staff reviewed numerous technical reports and topical programs related to system integrity, including the analyses of the PWS system structural integrity as documented in "Topical Report of Structural Integrity of the SRS Reactor Process Water System" (Reference 2), Reactor Materials Program reports (References 3 through 42), LBB Topical Report (Reference 43), and the Reactor Tank Ultrasonic Test Program.

• 6.7.3.1 Restart Criterion 1 -- Process Water System (PWS)

The WSRC program for validating PWS integrity addresses four major requirements:

- o Demonstrate compliance with applicable design codes and standards of record.
- o Establish the LBB capability and quantify margins of safety for the various PWS components.
- o Demonstrate the existence of a thorough ISI program to provide detect-before-leak capability.
- o Demonstrate ability to detect and respond to PWS leakage.

The components that constitute the PWS are as follows:

- o reactor tank and discharge nozzles
- o plenum and plenum nozzles
- o PWS piping
- o PWS circulating pumps
- o valves
- o PWS heat exchangers (HX)
- o PWS expansion joints
- o large bore piping through the first remotely operated valves in connected systems

Validation of the integrity of the PWS pressure boundary requires proper consideration of loads and resulting stresses; fabrication and modification process control; inspection and testing degradation due to aging, corrosion, fatigue, irradiation effects, and IGSCC; and replacements and corrective actions taken and/or being implemented.

The following subsections for this criterion discuss the staff's evaluation of the impact of these considerations.

Restart Criterion 1.a -- Engineering Analysis and Design for PWS

This restart criterion requires that WSRC demonstrate the structural safety margins stated in the Safety Analysis Report (SAR) are consistent with current plant configuration and will be maintained. These margins should be consistent with the national codes and standards pertaining to seismic (discussed in the seismic section of the SER) and other postulated design basis accidents.

The staff conducted a review of this criterion through the review and evaluation of available documentation, reports, and past and current program results. (References 44 through 58).

The following restart issues are part of the upgrade implemented through the incorporation of the "K-Reactor Restart Strategy, November 1988" (also known as the Charlotte Criteria) plan:

DB-2.3 - Seismic Resistance -- Ensure that the PWS is seismically qualified by analyses and by DOE approved walkdowns, including consideration of effects of failure of nonseismic structures (including the expansion ring), systems, or components on safety equipment. (See SER Section 6.1 for a detailed description.)

DB-6.4 - Vessel Integrity Status -- Review adequacy of technical information on vessel integrity status and results of previous inspections and analyses. By extension, this scope is expanded to include a review of the status of the remainder of the K-Reactor PWS and of the long-term plan aimed at ensuring the continued integrity of the PWS. WSRC-RP-89-152, Rev. 1 (Reference 44) provides a detailed summary of information available as of May 1989 relative to the structural integrity of the K-Reactor tank, including fabrication and operating history, results of nondestructive examinations, investigation of potential modes of material degradation, fracture mechanics and stress analyses, and results of Leak-Before-Break (LBB) evaluations. WSRC technical memorandum SRL-EDG-900274 (Reference 39) summarizes the results of the ultrasonic (UT) examinations of the K-Reactor tank which have occurred since May 1989, and which have concluded that no evidence exists of IGSCC in the K-tank welds.

DOE staff review of these two companion documents concluded that they provide an acceptable basis for the justification of the structural integrity of the K-Reactor tank in response to Reactor Operations Management Plan (Reference 58) Section 1.6, Restart Work Item DB-6.4. Also in response to Reference 65, WSRC-RP-90-099 (Reference 59) provided the basis for the reliability and capabilities of the PWS leak detection system, which is an important factor in the establishment of the LBB capability of the Process Water System including the reactor tank as well as the PWS piping and associated components. The PWS leak detection capabilities are addressed further in the section below dealing with Restart Criterion 1.e. The long-term plan for continued in-service inspection and testing of the Process Water System is addressed in SER Section 14.0.

WSRC is conducting a review of the SAR and will make a comparison of the codes and standards in the SAR with present analyses. WSRC has issued a report (Reference 40) that summarizes the structural margins for the as-designed condition of the PWS and identifies the codes and standards used for PWS design and fabrication. However, the DOE staff review indicated that it was not acceptable for closure of ROMP Restart Work Item SE-6.7A, because the required comparison to the SAR design basis was not included in the report. A report will be issued on the results of the comparison prior to restart. This is a WSRC deliverable (SE-6.7A) for Restart Criterion 1.a and is an open item.

WSRC programs related to the following topics will provide verification that the structural safety margins are consistent with those stated in the SAR and with appropriate national codes and standards.

Reactor Stress Analysis Evaluation

Compliance with the design rules of ASME Code Section VIII of 1949 and subsequent verification of the stress analysis and comparison with the current ASME Section III Class 1 design allowable stresses ensure that the tank stresses are within allowable limits and provide adequate safety margins. In another stress evaluation performed by Quadrex Corporation, ASME Section VIII, Division 2 rules and allowable stresses were used, and the stresses in the tank and the expansion ring were found to satisfy the applicable criteria.

Reactor Fracture Mechanics Assessment and Leak-Before-Break (LBB) Evaluation

Linear elastic fracture mechanics (LEFM) and elastic plastic fracture mechanics (EPFM) techniques were used to assess the reactor tank integrity. Using both methods, WSRC has made a direct comparison of a hypothetical crack length at instability to the crack length at the onset of stable crack growth, or at crack initiation. This analysis provided the crack length at the onset of instability for the various regions of the tank. The analysis was based on the ASME Code Section XI rules with a safety factor of 3.0 for normal operating conditions and safety factor of 1.4 for accident conditions. Material strength and fracture resisting toughness are the two major material properties used in fracture mechanics assessments. This is a long-term, continuing program.

The Safety Analysis Report (SAR) for the SRS production reactors addresses postulated accidents that are to be considered in the design of the reactors. One of the accidents considered is the abrupt double-ended guillotine break (DEGB) of a large process water pipe. In 1984, the Savannah River Laboratory (SRL) formulated the Reactor Materials Program (RMP) to provide a technical basis for the elimination of the DEGB as a plausible mechanism of pipe failure. This program, which made use of the leak-before-break (LBB) methodology, resulted in the generation of a summary topical report with 36 supporting technical reports (Emergency Cooling System-Loss of Cooling Accident [ECS-LOCA] Reports). Although the leak-before-break alternative to DEGB as a design basis has not been formally proposed by WSRC, the technical documents generated in support of the LBB initiative provide a detailed justification of the structural integrity of the Process Water System (PWS) including the reactor tanks.

One of the many design criteria and requirements for the design of commercial nuclear power plants is referred to in Title 10 of the Code of Federal Regulations, Part 50 (10CFR50) Appendix A, as General Design Criterion 4 (GDC-4). This criterion requires, in part, that structures, systems, and components important to safety be designed to accommodate the effects of postulated accidents, including loss-of-coolant accidents (LOCAs). However, dynamic effects associated with postulated pipe ruptures may be excluded from the design basis when analyses demonstrate that the probability of fluid system pipe rupture is extremely low under conditions consistent with the design bases for the piping.

The analyses that have been used by the commercial nuclear power industry, subject to NRC approval on a case by case basis, make use of the LBB concept to justify exclusion of the dynamic effects of postulated pipe rupture. This concept is based on the ability to detect a fluid system leak and perform an orderly and controlled plant shutdown before any potential exists for catastrophic pipe failures. Thus, the object of applying the LBB concept is to establish that a postulated crack remains stable under normal operating plus faulted loads or that significant margin exists against unstable crack growth if the postulated crack is predicted to grow with the applied loads. An unstable crack would continue to grow under a constant applied load without arrest until complete fracture of the pipe occurred.

Acceptance criteria for this methodology have been published by the NRC staff as NUREG-1061, Volume 3 (Reference 60). The same criteria are also contained in the NRC Standard Review Plan, Section 3.6.3, "Leak-Before-Break Evaluation Procedures." WSRC/SRL has made use of the NUREG-1061 acceptance criteria in the establishment of the Reactor Materials Program for detailed evaluation of the LBB concept for the SRS reactors. These criteria are listed in the SRL Summary Report (Reference 61), and the responses to these criteria are summarized in Reference 4. WSRC has also prepared two documents summarizing the information contained in the ECS-LOCA reports. WSRC-RP-89-236, Draft Topical Report, "Leak-Before-Break Capabilities and Related Safety Implications for the Savannah River Production Reactors" (Reference 43), and WSRC-RP-90-559, Topical Report, "Structural Integrity of the Process Water System (PWS) of the Savannah River Site Production Reactors," dated June 30, 1990 (Reference 2), provide a detailed summary of the technical justification of the structural integrity of the PWS pressure boundary. All of the NUREG-1061 acceptance criteria have been addressed in the WSRC technical documentation. It is recognized that all of the criteria required for applicability of the LBB concept have been explicitly met, with the exception of the requirement that the operating history of the system materials show no evidence of IGSCC. Although IGSCC has been observed in the heat-affected zone of some PWS piping circumferential welds, this alone does not invalidate the use of the LBB fracture mechanics methodology to establish the structural margins of safety required to demonstrate the structural integrity of the PWS.

LBB evaluation at SRS included both probabilistic and deterministic analyses to demonstrate that the likelihood of a significant pipe break is acceptably low. Although deterministic methods cannot be used to directly obtain failure probability, reasonable component failure probabilities were established using applicable historical data. Allowances were made for uncertainties and the

possibility of significant loads or in-service degradation by requiring margins on failure-related variables, such as flaw size and load. The tank fracture assessments used nominal and lower bound estimates for the material properties and took into consideration thermal as well as radiation effects.

A major portion of this work has been completed, and the results have been presented in a series of technical reports (References 3 through 38). The application of the NUREG-1061 LBB criteria to the SRS PWS will represent a very conservative analytical approach, because these criteria specifically address failures postulated in high energy piping systems as opposed to moderate energy piping systems. The PWS maximum normal operating pressure at any power level and low stored energy resulting from the incompressible fluid flow conditions justify treating the PWS as a moderate energy system with respect to the NUREG-0800 (Reference 62) requirements for postulation of an instantaneous, double-ended pipe rupture. A moderate energy system, according to NUREG-0800, does not require postulation of instantaneous double-ended guillotine pipe breaks (DEGB). The analysis of the PWS, in accordance with NUREG-0800 and NUREG-1061 criteria, is, therefore, a more comprehensive and conservative approach than required by those documents.

In summary, the LBB methodology will be continuously developed by WSRC through issuance of various topical reports that apply to SRS reactors. DOE staff has completed the evaluation of the WSRC responses to its earlier set of 95 comments on the WSRC application of LBB methods. A formal acceptance of the WSRC responses has been issued. The staff has also reviewed the WSRC topical report on the structural integrity of the PWS (Reference 2). The review has concluded that this report provides an acceptable summary of the technical basis for PWS structural integrity based upon the methodology and acceptance criteria of NUREG-1061, Volume 3.

Degradation Mode and IGSCC

The PWS has been susceptible to intergranular stress corrosion cracking (IGSCC) and irradiation-assisted stress corrosion cracking (IASCC). The effects are assessed below.

IGSCC

Laboratory tests have been conducted under conditions that simulate the moderator chemistry. IGSCC occurred in all tensile specimens made from the Type 304SS pipe taken from the R-Reactor. The specimens were pulled at a slow constant rate at 105 degrees C under water to duplicate moderator chemistry. IGSCC is aggravated by increased chloride and sulphate concentration. Although IGSCC has been observed in the heat-affected zone (HAZ) of about 30 circumferential pipe welds in the process water piping of the K-Reactor, no leaks from the K, L, or P-Reactor tanks have been observed since the original startup. The C-Reactor tank leaked in the knuckle regions, but K, L and P-Reactors do not have knuckle regions. Visual inspections of the tank wall has not revealed any cracks. Ultrasonic Testing (UT) of the P- and K-Reactor tanks has been successfully completed, with no IGSCC found. For detailed descriptions of the reactor tank UT results refer to SER Section 14.0, "Inspection and Testing."

A significant crack-like indication in one of the K-Reactor 16-inch diameter piping welds in the PWS was reported by WSRC in early August 1990, as a result of the piping reinspection program. This indication had been previously identified as a welding defect, but was subsequently characterized as a flaw potentially resulting from IGSCC. However, the extent of crack "growth" (if any) during the early stages of the K-Reactor outage was indeterminate. Because the subject flaw exceeded the allowable crack size specified in the pipe replacement criteria, it was decided to remove the section of piping containing the flaw and conduct metallurgical evaluations and destructive testings to better characterize the UT examination results. A structural evaluation of the flaw was also conducted by WSRC (Reference 63). It concluded that the flawed section would have remained stable and that there was adequate margin for combined normal and seismic loading in accordance with ASME Section XI criteria (Reference 64). Subsequent WSRC metallurgical evaluations (Reference 65) of the actual removed piping section concluded that the reported indication was not a flaw and that there was no evidence of IGSCC or cracking of any type. The UT indication was a result of the reflection from the intersection of the grain boundaries of two overlapping full penetration, vee-groove welds resulting from the installation of a piping spool piece to accommodate proper fitup during the original piping installation.

During the April 1990 and October 1990 Defense Nuclear Facility Safety Board (DNFSB) meetings on System Integrity, the Board asked for rationale for the absence of cracking in the reactor tank welds, which were exposed to the same environment and were made of the same material as the PWS piping, which has shown indications of IGSCC. WSRC responded with the rationale that for IGSCC to occur the following three conditions should be present: (1) the alloy is sensitized, (2) a corrosive environment exists, and (3) sufficient tensile stresses are present. These conditions differ in the PWS piping welds in comparison to the reactor tank welds.

A WSRC technical evaluation was initiated to address the factors that account for the differences in IGSC cracking tendencies between the PWS piping and the reactor tanks. The results of the WSRC evaluation (Reference 66) identified a number of dissimilar characteristics between the piping and the tanks including fabrication methods, welding techniques, base material composition, and service-induced irradiation. The effects of these differences on residual tensile stress levels and on the degree of sensitization of weld heat affected zones provide the rationale for the lack of IGSCC in the reactor tank welds compared to the PWS piping welds.

The fabrication and welding procedures used for reactor tank construction lead to the conclusion that residual tensile stresses in the tank welds are less severe than in the piping welds. Procedures for tank shell fabrication included welding flat plate sections using a double-bevel V-groove weld profile with weld metal deposited from both sides. These plates were then roll-formed to the cylindrical shape required for the tank. On the other hand, piping fabrication included single-bevel, V-groove welds with weld metal deposited from the outside only. The symmetrical weld profile with subsequent mechanical stress relief due to roll-forming of the tank provides for less severe residual stresses in the tank welds, and, therefore, a reduced IGSC

crack driving force when compared to the piping welds. Residual stresses in the tank shell, especially in the mid-plane elevations, are further reduced by irradiation induced stress relaxation; thus, further reducing the susceptibility of the tank shell welds to IGSCC. Reference 76 also concluded that the low carbon content of much of the tank shell plate material decreased the possibility of sensitization of the tank material during welding. The combined effects of these differences in fabrication procedures, service-induced irradiation, and material characteristics reduce the tendency for initiation of IGSCC in the reactor tank welds vs. the PWS piping welds.

The staff concluded that, in view of the past experience and the results of the augmented reactor inspections, there is no indication of service-induced degradation (IGSCC) in the reactor tanks.

Irradiation Assisted Stress Corrosion Cracking (IASCC)

Irradiation-assisted stress corrosion cracking has been identified in solution-annealed Type 304 stainless steel (SS) that was irradiated to high fast neutron fluences at temperatures around 288 degrees C. Irradiation enhances grain boundary segregation of impurities that are normally present in the SS; therefore, IASCC is of concern to sensitized austenitic SS present in the reactor tanks. However, the reactors at SRS operate at temperatures less than 120 degrees C, and at this relatively low temperature vacancy diffusion rates are too low for grain boundary separation to occur. Therefore, the staff concluded that IASCC is not a concern in the SRS tanks.

Transgranular Stress Corrosion Cracking (TGSCC)

Other PWS components, such as nozzles, plenums, piping, pumps, heat exchangers, valves, expansion joints, etc., have design characteristics that affect their susceptibility to in-service degradation. Their susceptibility is also affected by the chemical composition and thermal-mechanical history of the steel and the stresses (both applied and residual) acting on the steel. One observed form of degradation to such components is transgranular stress corrosion cracking. This type of degradation has caused leaks in the heat exchanger tubing and heads, tank effluent nozzles, and large diameter piping. The cracking is frequently initiated on the outside of pipes and is assumed to have been promoted by chloride ions, which were leached from neoprene and other synthetic rubber compounds, paints, inks, insulations, and labels. Elimination of higher level of chloride containing material from service should prevent this type of degradation.

Fatigue

Some cracks in heat exchangers, expansion joints, small diameter (6-inch and under) piping, and small-diameter branch connections (2-inch and under) have been attributed to fatigue. The failure of the large PWS expansion joints was primarily associated with excessive vibration-induced fatigue following the installation of the Process Water Pumps in the late 1950s. This problem was subsequently eliminated by a design change in the expansion joints. All the small expansion joints in the K-Reactor have been replaced with an improved design that uses 2-ply bellows. The 2-ply feature of the new design allows

for inspection by a hydrotest in between the bellows. These bellows are covered by a wiremesh screen for protection from external damages. They also feature an improved design of the flow sleeves. (See the section on Process Water Expansion Joints for more details on the design of the expansion joints.)

Plenum Nozzles and the "J" Bolts

The reactor inlet plenum nozzles are restrained by roller anchors, which are fastened to the floor by "J" bolts. Under postulated accident conditions such as a loss-of-pumping accident (LOPA), the pressure inside the reactor and the resulting forces on the combined structure of the plenum, top shield, six inlet nozzles, and the forest provides uplift forces that are resisted by the roll anchors and "J" bolts. The "J" bolts are made of ASTM A283, Grade C material with a yield strength of approximately 30,000 psi.

If the roll anchors fail, the plenum could lift and stretch the plenum seal, which then may rupture. The rupture would provide pressure relief to protect other components. The roll anchors, plenum inlet nozzles, and plenum seals were not originally designed specifically for overpressurization of the gas plenum. However, modifications have been made to the roll anchors to prevent uplift of the plenum in the event of an ECS actuation, and they are now adequate for the uplift force due to gas plenum overpressurization (Reference 49).

Piping Design and Fracture Resistance Analysis

Archival piping material taken from the retired SRS reactors has been tested for mechanical properties. The tests have provided baseline data for irradiated and corrosion-prone material specific to the PWS service conditions. The data (Reference 53) indicated that the stainless steel remained tough and ductile under the operating conditions.

The PWS piping system was designed and built to ANSI B31.1 Code requirements. All full-penetration welds were inspected by radiography and, upon installation, a hydrotest was performed to validate the design parameters. A stress analysis based on ASME Section III, Class 1, was successfully performed for the piping system. Fracture toughness of the piping material was evaluated by the WSRC Reactor Materials Program (RMP) analysis of the J-R curves obtained by testing of specimens conforming to ASTM E813-81. The results indicated that all fractures were ductile at both 25 degrees C and 125 degrees C for the base metal as well as weld material. Impact tests made at 25, 75, and 125 degrees C indicated high toughness of the 304SS. Toughness remained high for the temperature range of operation of the piping system for base, weld, and HAZ material for all orientations.

Heat Exchanger Design Bases, Fabrication, and Modifications

The PWS heat exchangers (HX) are designed to remove the heat generated in the reactor core and transfer the heat to cooling water on the shell side. The original HXs were designed and fabricated to the rules of ASME Code Section VIII of 1949, but no code stamp was applied. The replacement HXs were

designed to the ASME Code Section VIII, 1980 edition through the summer addenda (no code stamp applied) and to the Tubular Exchanger Manufacturers Association (TEMA) Standards of 1968. The tubes are of ASTM A249 and A450 specification, bead-conditioned, full-finished, annealed and pickled. The tubes are cored with SS filler rods to produce a narrow flow annulus on the primary coolant side.

The HXs are supported on wheels to permit easy removal and replacement. The wheel under-carriage structure is seismically supported. The maximum pressure difference across the HX tubes is about 115 psig. Any leakage that occurs will be from the PWS side to the cooling water system. The principal mechanisms for leak initiation are corrosion, erosion, vibration fatigue, and cracking of the stainless steel tubes. The release of radioactivity to the environment that would result from postulated leaks is small, because the leak rates are small. Therefore, failure of the tubes would not cause a serious LOCA in the PWS. Furthermore the cooling water discharge is monitored by gamma monitors to detect leakage from higher pressure PWS. An analysis by WSRC (Reference 9) has demonstrated the pressure boundary integrity of the HX shell. The analysis also addresses the very conservative postulation of failure of all the HX head staybolts and shows that the HX C-clamps provide backup load carrying capacity in the event of staybolt failure. The HX staybolts will be subject to an ISI Program, which is being developed to monitor the integrity of the staybolts.

In summary, the integrity of the HXs are affected by degradation due to corrosion, erosion, vibration-induced fatigue, and cracking of the stainless steel tubing. Leakages, which were small in magnitude, have resulted in the past due to such degradation. Technical Specification Section 4.2 states that the moderator flow carried by each tube averages 1.4 gpm and is considered to be small. The tritium leaking into the process water is detected by radiation monitors. If the reactor operation is unaffected by the leaks and the radiological releases are within acceptable limits, the operation can continue. Otherwise, the reactors will be shutdown, leaks located, and repair or replacement of the faulty equipment shall be initiated without delay. In addition, the original HXs have tubes of Type 304SS material; however, the replacement tubes are of Type 316SS material, which is expected to be more resistant to corrosion.

Process Water Expansion Joints

Two different sizes of expansion joints (EJ) are used in the PWS system (small and large nominal diameter sizes). The small EJs are attached to the primary side outlet connection of each of the 12 PWS-HXs located in 6 primary loops of each reactor. The large nominal-size EJs are located on the suction side of each of the Process Water Pumps, between the suction inlet and the isolation valve. The fundamental difference between the two sizes of EJs is in the type of end connections.

A total of 2200 expansion joint-years of history has been recorded at the SRS, and a total of 19 minor leaks occurred in this period. Fourteen of these leaks were in the large-size EJs, and five were in the small sizes. Apart from these leaks, three tie-rod failures occurred due to failure of the nuts.

Some of the leaks were due to crevice corrosion. Painted spacer rings have been determined to be a source of chloride contamination and corrosion of the SS bellows. The K- and L-Reactor EJs were inspected thoroughly under a program undertaken by the operating contractor. Ultrasonic thickness inspections for the small sizes and a special profile radiographic technique for the large sizes revealed no thinning of the flow sleeves.

The expansion joints in the PWS system have been identified as the most probable location for a significant break. The LBB analysis has focused on various details of construction of the expansion joints. One specific area is the annular gap area between the free end of the annular sleeve's outside diameter and the inner diameter of the stub end pipe where the bellows end is welded. In the event of a postulated failure of the bellows, this annular gap would provide the maximum possible flow path, assuming the sleeve remains in place. The maximum leakages past this gap have been closely analyzed by the LBB method under the postulated accident scenarios. The basis of this determination is linked to the troubled operational history of the expansion joints, the damages sustained, and other vulnerabilities described above.

It has been postulated that a rupture in the large expansion joint would result in a 4000 gpm leakage and that a rupture from a small joint would yield a 3000-3500 gpm leakage. This is significantly less than a postulated 60,000 gpm leak which would result from a medium-size-diameter line break; therefore, the consequences of a bellows joint failure are much less severe and result in a lower rate of water level drop in the reactor tank. This slower rate provides enough time for the reactors to be scrammed. A bellows joint failure also permits the ECS system to preclude the ingress of air and preserve a high level of coolant inventory in the latter stages of the accident.

In consideration of the weaknesses in the design and construction of the existing joints, a replacement and upgrade program is currently underway at SRS to improve upon their reliability and integrity. The following restart item is applicable to this subject.

Outage Work-9 (OW-9) - Replace Small Process Water Expansion Joints -- Expansion joints on the process water system will be replaced as part of the Leak Before Break evaluation."

In order to meet this requirement, WSRC has replaced all the small expansion (12-inch diameter) joints in the K- L-, and P-Reactors with an improved design of 2-ply bellows. The 2-ply feature of the new design allows for inspection by a hydrotest in between the bellows. These bellows are covered by a wiremesh screen for protection from external damages. They also feature an improved design of the flow sleeves. For the lower pressure, large expansion joints, the results of their inspections indicated that they are acceptable for continued operation and, therefore, replacement of the large expansion joints is not a restart issue. However, the large PWS expansion joints will be replaced in the longterm with a modified design.

In summary, the PWS expansion joints are currently subject to inspections at every 5-year interval per existing ISI programs and will be subject to augmented and upgraded inspections per the new ISI programs being implemented.

Based on completion of the replacement program (OW-9) and the implementation of the upgraded inspection program, the staff finds the expansion joints for the PWS to be acceptable.

The Process Water Pumps and Valves

The reactor PWS pump is a horizontal, single-stage, double-suction, double-volute, vertically split centrifugal unit that has a large suction and a medium discharge line. The pumps have double mechanical seals on each side of the shaft. The pump casing and the impeller are cast CF-8 SS, and the shaft is made of 17-4 PH SS. The PWS pumps are made of corrosion resistant SS material with backup drive sources. The pumps are subject to periodic disassembly and internal inspection and testing according to the current and upgraded ISI plans (also refer to SER Section 14.0). The pump bodies have much stronger construction than needed to maintain structural integrity and are not susceptible to IGSCC. The flywheel material per Technical Specification Section 2.2 is ASTM A284-49T, Gr. B, open-hearth steel plate, heat-treated to refine grain. The pump flywheel integrity is ensured on the basis of a sound design, material selection, and thorough inspections.

The valves are designed and fabricated to operate under a wide range of operating conditions and have a much higher margin of safety than that required to maintain structural integrity for the design conditions. The valves are made of cast stainless steel and are not susceptible to IGSCC.

Restart Criterion 1.b -- Fabrication and Modification Process Controls for PWS

This criterion requires that fabrication process controls must be exercised during all stages of component manufacturing and system modifications to avoid severe sensitization of the material and to minimize exposure of the stainless steel to contaminants that could lead to stress corrosion cracking.

During the original fabrication period in the 1940s, conventional methods were used for the reactor tanks. All welding was done in accordance with the rules of the ASME Code Section VIII, 1949. As stated earlier, austenitic stainless steel was used in the construction of the reactor tanks; therefore, process controls were exercised during all stages of component manufacturing and reactor fabrication to avoid sensitization of the material. In addition, the weld metal composition and the welding procedures were properly controlled to prevent hot cracking (microfissuring) of the SS during welding. The original PWS design and fabrication were controlled through specifications to minimize stress corrosion cracking of the stainless steel. However, the phenomenon of IGSCC was not entirely known 35 years ago when the reactors were built, and some of the components were inadvertently sensitized during the fabrication process. The major past practices, now eliminated, that caused sensitization were as follows:

- o Flame washing, i.e., the use of flames to heat up piping ends to aid in the match up of field joints, and other heat treatments that were made to the Type 304SS without subsequent solution heat treatment.

- o Pickling of Type 304SS in a mixture of nitric acid and hydrofluoric acid to remove scales and other contamination produced during the fabrication process.
- o Contamination of the SS material with chloride-containing material, such as cleaning agents, solvents, neoprene gaskets, and masking tape.

Corrective procedures have been implemented to minimize, or eliminate where possible, the above-mentioned sensitization by implementing the following measures.

- o Replacement piping is of a lower carbon grade of stainless steel, Type 304L or, 316NG (0.02 percent Carbon), which is less prone to sensitization.
- o Grit blasting methods are used instead of acid pickling.
- o Flame washing has been eliminated.
- o Chloride content of paints, markers, etc. has been reduced to acceptable limits.
- o Improvements in the methods used for stress relief of field welds have been investigated, and improved heat sink welding methods will be used at SRS. This method has demonstrated the reduction of residual stresses in field welds. The effectiveness of these welding procedures was verified by the presence of compressive stress on the inner surface of piping.

The staff concluded that through the elimination of these past practices and implementation of the corrective procedures mentioned above, compliance with Restart Criterion 1.b has been established. WSRC has issued a report (Reference 68) that provides a summary of PWS fabrication procedures, identifies past practices that may have contributed to occurrences of IGSCC, and indicates how current procedures control the equipment fabrication and modification process to minimize IGSCC.

Restart Criterion 1.c -- Water Chemistry for PWS

This criterion requires that Primary System water chemistry shall be controlled by preventing contamination and through purification to prevent IGSCC. Procedures shall be established to ensure that chemical parameters are controlled within the limits specified in the SAR.

Moderator chemistry maintenance requirements, limits, procedures, past and present practices; proposed upgrades during plant operation and shutdown (lay-up) phases; and future improvement plans are discussed here. WSRC has prepared a series of reports on moderator water chemistry (References 54 through 57 and 67), which identify the requirements for monitoring, sampling, recording, and evaluating PWS moderator chemistry data. The requirements for PWS chemistry control during startup, charge and discharge operations, and

shutdown are summarized, and details such as moderator chemical composition limits are specified. Reference 67 also provides an implementation plan for incorporation of recommended enhancements of the SRS moderator chemistry program and categorizes these recommended improvements as post-restart issues that have no direct impact on safe operation of the reactors. The DOE staff reviewed References 54 through 57 and 67 and concluded that these documents satisfy the ROMP Restart Work Item SE-6.7C requirements for monitoring, sampling, recording, and evaluating water chemistry data.

For this restart criterion, the staff concluded that the PWS moderator chemistry is monitored and controlled with proper care within acceptable limits and that programs have been developed to self-examine the current practices and controls for improvement of the current state of the art. The DOE staff reviewed the WSRC moderator chemistry program documentation and concluded that it is sufficient to satisfy the requirements of Restart Criterion 1.c.

This results of the staff review of this restart criterion are discussed below.

Chemistry During Plant Operation

The chemistry of the heavy water (D_2O), which serves as the coolant and the moderator, is maintained to control corrosion of both the Type 304SS material of the reactor tank and piping and the aluminum cladding of the fuel and the target assemblies in the reactor tank. Corrosion of the reactor components may produce fuel cladding failures, leaks in the hydraulic system, and excessive turbidity and radioactivity in the reactor systems. A secondary objective of the chemistry control is to minimize decomposition of the heavy water during reactor operation. It has been demonstrated that such corrosion problems are substantially reduced by the control of the acidity of the moderator. The quality of the moderator is specified in the Technical Standards and Operating Limits, Table 7.6.2.1.3.1. No significant decomposition of moderator has been experienced at the SRS. This is attributed to recombination of free radicals formed by radiolysis in the high gamma fields present in the reactors.

Continuous on-line measurements of conductivity, pD, and dissolved oxygen, are supplemented by routine grab sample analyses. Samples are taken at specified intervals varying from three times a week to once a month, depending on the results of the analyses. The results are stored in a database for trending purpose. Knowledge of the moderator chemistry has been augmented by special studies.

The present water chemistry is a balance between minimum stainless steel cracking and a low aluminum corrosion. To protect the aluminum fuel cladding, the moderator is maintained slightly acidic (pD between the limits of 7.4 and 4.4 per Reference 62 during reactor operations) with nitric acid, with oxygen added to stabilize the nitric acid. To protect the SS in the PWS, the added oxygen is minimized (0.5 ppm), and the deleterious chlorides are removed by recirculation through mixed-bed ion exchangers. The conductivity of the moderator is maintained at no greater than 3.3 mho/cm as measured at 25

degrees C when the reactor is in normal operation. The rise of conductivity to a higher limit of 5.0 mho/cm is allowed for a day, while corrective actions are taken to bring the conductivity to the normal limits.

A comprehensive study of the moderator quality by WSRC confirmed that contributors to the IGSCC problems of the SS in the PWS are temperature, hydrogen peroxide, and the combined effects of chloride and sulfate ions at the nominal concentrations of 30 and 50 ppb, respectively. (Reference 56)

Chemistry During Lay-up

The K-Reactor was shutdown in April 1988 and the fuel discharged. The tank was drained in August 1988 and remained empty till early December 1988. When the Stop Work Order was issued in late December 1988, the moderator level in the tank was 13 feet and remained in that level till mid-April 1989. Overflow circulation has been maintained since late April 1989. Between April 1988 and April 1989, no routine analyses of K-Reactor moderator were made, and the condition of the SS and aluminum components in the tank was unknown. Upon an analysis in April 1989, pD, conductivity, nitrate content, chloride content, sulfate content, turbidity, and metals analyses were all found to be at a higher limit than the operating limit. Because of this deviation, DPSOL-6133, Rev. 2, "Guidelines for Corrosion Mitigation during Protracted Reactor Shutdown" (Reference 76), has been issued and implemented. This document is the procedure for control of water chemistry during the current lay-up.

Moderator circulation was provided by heavy water addition and overflow. One purification cell was put in operation in May 1989. Since then, the conductivity and pD of the moderator has returned to within normal operating range. Operation of the Process Water Pump DC motors was started within a short period. The total flow was about 90 gpm with about 1/3 of the flow being purified. Sampling was available only in purification at that time. Metallurgical analyses by WSRC concluded that the changes in the K-Reactor moderator chemistry effected by the 1988-89 shutdown did not cause a significant increase in the corrosion of austenitic stainless steel of the reactor tank and the PWS piping system. (Reference 56)

Upgrading of the Moderator Chemistry

WSRC Reactor Engineering Department (RED) has developed a Moderator Chemistry Program for upgrading the moderator chemistry. The objectives of this program are as follows.

- o Document and provide the technical bases for the methods, systems, procedures, requirements, and administrative controls currently used at SRS to maintain, control, monitor, analyze, and record moderator quality during operational and lay-up phases.
- o Examine the adequacy of such procedures, requirements, and administrative controls against the state-of-the-art practices in heavy water chemistry, and the applicable commercial industry practices and guidelines.

- o Develop a timely and comprehensive implementation plan for the identified items for improvement in the current methods, systems, and administrative controls applicable to moderator chemistry.
- o Implement the plan.

Restart Criterion 1.d -- In-Service Inspection (ISI) for PWS

This restart criterion requires (1) the structural integrity of the material to be demonstrated and maintained by implementation of an ISI program, (2) NDE to be performed to demonstrate the reactor tank integrity, and (3) administrative controls and procedures to be developed and implemented to provide guidance on repairs/replacements of components.

In-service inspections of the reactors date back to 1963. In 1986 a visual inspection was performed remotely, covering 100 percent of the tank welds on the inner surface of the tank. Visual inspections performed to date showed no signs of degradation or surface cracks in the K, L and P-Reactor walls (Reference 2). P- and K-Reactor UT inspections have been completed and have not revealed any cracks. L-Reactor tank UT is progressing (as of December 1990). The UT method for the tank examination used state-of-the-art techniques similar or superior to those used in the commercial nuclear industry. The inspectors are properly trained and certified by the Nondestructive Testing Center of the Electric Power Research Institute (EPRI) Center in Charlotte, NC.

All accessible circumferential welds in the PWS pressure boundary piping, 3 inches and larger, have been examined by UT. This included the septifoil coolant piping and the SS portion of the ECS piping. All piping containing welds with UT crack indication greater than the approved restart criteria (50 percent of wall thickness in depth or 20 percent of the circumference in length) have been replaced; and welds with any indications have been placed on an increased frequency monitoring program.

Restart Criterion 1.e -- Leak Detection for PWS

This restart criterion requires that justification for a sensitive leak detection capability should be established to detect specified small leaks and that leak-detection capability should be consistent with Technical Specifications that specify the operability of leak detection equipment.

WSRC has issued a summary report (Reference 42) that provides the basis for the capability, sensitivity, and reliability of the PWS leak detection system. As a result of the evaluation of this criterion, the staff concluded that there is reasonable assurance that online leak detection system will provide large margins of leak detection capability for SRS reactors. This conclusion is based on the fact that multiple diverse backup systems are installed to detect leaks. Procedures are also implemented to routinely detect leaks. The following is the evaluation of this criterion.

The presence of tritium in the moderator, the stack tritium monitors (STM), and the Berthold Tritium Monitor (BTM) provide a very sensitive means of

detecting leakage. These monitors can selectively sample airflow from certain reactor areas; the losses from all process areas can be monitored at the reactor building stack. Against fluctuating normal background losses, the monitors can reliably detect leaks as little as 10 pounds/day. Actual leaks of approximately 20 pounds/day are considered normal process losses. The operating procedures require that at least two STM/BTMs be online during reactor operation. Backup to the STM/BTMs is provided by an additional leak detection system. The Kanne system can selectively sample from specific locations of the reactor process area for tritium. Technical Specifications and operating procedures have been upgraded to ensure continuous online leak detection capabilities. Operator actions are defined in these procedures for leak rate exceeding 50 pounds/day. In addition to the monitors, two auxiliary systems are provided for visual detection of the leaks. Closed circuit television cameras are located throughout the building and are monitored in the reactor control room.

The following action item is part of the upgrade implemented through the incorporation of the "K-Reactor Restart Strategy, November 1988" (Charlotte Criteria) plan. L-Reactor has been equivalent program.

DB 2.4 - Technical Specifications/Standard on Leak Limit Detection -- Issue a technical specification on leak limits and leak detection capability, including limiting conditions for operation, contingency actions, surveillance requirements, and technical bases.

The leakage limit in the technical specification is based on SRL calculations of unstable crack length for the PWS and/or historical data. Technical Specifications 3.4.1 and 3.4.2 contain the information required by Charlotte Criteria DB 2.4 on leak detection capability and limiting conditions for operation.

Summary for Evaluation of PWS

Based on the above review for the Restart Criterion for PWS, the staff found that the PWS, including the reactor, piping system, and the essential components satisfy their original design requirements and fabrication standards and codes. The inspection of the reactor tanks (P and K) UT has been completed, and the integrity of the tanks has been demonstrated. The PWS pressure boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimal. The staff concludes that the pressure boundary integrity of the PWS, including the reactor tank, main process piping, the septifoil supply system piping, pumps, and bellows joints, has been demonstrated and provides reasonable assurance that the reactors will function safely.

6.7.3.2 Restart Criterion 2 - Engineered Safety Feature (ESF) Systems

This restart criterion requires that WSRC demonstrate that ESF systems have a low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The fluid and material compatibility for the ESF systems, and for the auxiliary systems that directly support the ESF systems, should be considered to reduce or eliminate the propensity of contaminant-induced piping

and component degradation. The ESF systems include the Supplemental Safety System (SSS), the Moderator Recovery System (MRS), and the Emergency Cooling System (ECS).

Evaluation of this criterion is given in the following sections.

Emergency Cooling System (ECS)

The ECS system is designed to remove decay heat following reactor shutdown when the normal decay heat removal system is not operable. The ECS provides for direct addition of light water to the reactor core to protect against the consequences of a loss of primary coolant (LOCA) accident and/or loss of pumping accident (LOPA).

Engineering Analysis and Design for ECS

The ECS is designed to prevent fuel melting for worst case single failure accidents. Full ECS flow to the reactor is required within 20 seconds after an actuation signal is received or when plenum inlet pressure reaches 90 psi (at approximately 20 seconds for a large break LOCA). Seismic qualification of the ECS is addressed in the Seismic Section of the SER.

The maximum pressure developed by the ECS pumping sources is limited by design so that injection cannot take place against the reactor inlet plenum pressure when the PWS pumps are driven by the AC motors. This limits the shut-off head to approximately 70 psig and ensures against spurious light water injection. Maximum ECS flow is also limited by design and administrative control so the reactor vent capacity is not exceeded, which could overpressurize and lift the inlet plenum.

ECS piping was designed to DuPont Specification 3018, which (approximately in 1960) was superseded by DuPont Specification 4482. Codes contained in these specifications included ANSI B31.1, B31.3, B16.5, and ASME Section VIII. Piping and components connected to the PWS are made of type 304SS, and the recent replacement piping are of type 304L SS. Pump fluid temperature range is 40-90 degrees F, and the pumped fluid is river water filtered through a 1/8-inch-square mesh screen. The piping design pressure is 125 psig at 200 degrees F.

Fabrication and Modifications Process Controls

As described earlier, the ECS system was designed and fabricated to DuPont Specification 3018 (which included ANSI B31.1, B31.3 and ASME Section VIII) in the late 1940s and early 1950s. The carbon steel portion of the ECS, upstream of the last isolation valves to the PWS, is fabricated of ASTM A285 carbon steel plates and ASTM A53 plates. Thus, the ECS piping contains longitudinal seams in addition to the circumferential seams. Investigation has been initiated to review the fracture resistance of the ECS and Cooling Water System (CWS) piping to determine the adequacy of the piping material fracture toughness in the presence of cracks under functional operating loads. RIMP

Issue 780 was assigned to evaluate this concern prior to restart. ROMP Restart Work Item SCN No. 0214, discussed under the summary below, addresses this issue.

The following ROMP items are applicable to this Restart Criterion and shall be completed prior to restart.

WISR Item 6 - The ECS pumps should be monitored and/or tested periodically for flow degradation, minimum acceptance criteria shall be established, and the pumps shall be overhauled if necessary. This item is addressed in SER Section 14.0, Inspection and Testing.

WISR Item 39 - Provide increased surveillance testing at quarterly intervals where possible for the ECS injection lines. This issue was completed (Reference 68)

DB 6.1 - Baseline Walkdowns -- Establish on representative sample of key safety systems by Technical Baseline Program (TBP) walkdowns that the as-built conditions correspond to the safety analyses. This item has been completed. Refer to SER Section 5.3 for a detailed evaluation of this item.

Water Chemistry

The water source of the ECS is the river water contained in the 186 Basin. The river water is suctioned through grating to remove debris and other large solids. The retention in the 186 Basin allows the sediments to settle down at the bottom of the basin, which is cleaned periodically. The river water is chlorinated to minimize biofouling. The water is strained at the polybor header to remove smaller solids. The chemistry of the liquid in the header is maintained within limits to maintain the required boron poison concentration. Other than above, the river water is not purified, treated, or maintained to any specifications.

In-Service Inspection/In-Service Testing (ISI/IST) and Leak Detection

The ECS system is subject to appropriate periodic pressure and functional testing to ensure (1) the structural and leak tight integrity of its components; (2) the operability and performance of the active components of the system; (3) the operability of the system as a whole; and (4) under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, the transfer between normal and emergency power sources, and the operation of the associated cooling system. The SRS design and procedures provide for periodic testing of water delivery through the last isolation valve between the ECS and the PWS. In order to prevent moderator degradation, actual delivery of the light water to the reactor is not practical.

Summary for ECS

The WSRC programs, including WISR and RIMP issues for ECS, are in place to address Restart Criterion 2 for ECS. Adequacy of the fracture toughness of

the ECS carbon steel piping in the presence of cracks under functional operating loads was examined. WSRC has issued a report (Reference 69) in response to ROMP Restart Work Item SCN No. 0214, which provides a fracture mechanics assessment of the carbon steel piping in both the Cooling Water System (CWS) and the Emergency Cooling System (ECS). This report is being reviewed by the DOE staff and is an open item.

WSRC has also issued a report (Reference 70) in response to ROMP Restart Work Item SE-6.7F, which summarizes the status of the structural integrity of the ECS. This report is being reviewed by the DOE staff and is an open item.

Moderator Recovery System (MRS)

The moderator recovery system (MRS) is intended to supply reactor coolant makeup for small breaks in the PWS pressure boundary. The system safety function is to ensure that specified acceptable fuel design limits are not exceeded as a result of primary coolant loss due to leakages from the PWS pressure boundary and rupture of small piping or other small components which are part of this boundary. If light water systems are leaking concurrent with a PWS small break/leak, the MRS may pump degraded moderator into the PWS.

The system is designed to ensure that for onsite electric power operation (assuming offsite power is lost), and for offsite power operation (assuming onsite power is lost) the system safety function is accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation. This criterion has been met in the past for small breaks with less than 5 gpm leakage. If a break exceeded the low capacity of the D₂O makeup system, a reactor scram was called for, with possible ECS injection as a followup. To avoid this undesirable effect of ECS injection, a system with larger capacity and the potential for reinjection of the leaked moderator has been installed. This has special applicability to an Anticipated Transient Without Scram (ATWS) or seismic event combined with a PWS leak when SSS injection of soluble neutronic poison is used to shutdown the core. The MRS will help preserve the inventory of the gadolinium nitrate poison.

Engineering Analysis and Design

Since 1953, systems have been modified and new systems added to the SRS reactors as follows:

- o Systems and portions thereof added recently have been evaluated to the nuclear standard for Class 2 components, ASME Section III-NC. These include the MRS and the SSS.
- o The MRS is designed with a storage tank, which is the source of the moderator when the system is activated. The system is also designed for gravity injection to the reactor tank from the storage tank.
- o The MRS is designed to function in the event of a single active component failure. The system is designed to function upon the loss of offsite power considering a single failure. System piping

and the storage tank are designed to withstand the design basis earthquake (DBE). The system is functionally classified as seismic Class I and ANS Safety Class 2 (Reference 71).

The following restart issues are applicable to the MRS:

WISR-5 - Moderator Recovery System -- The MRS should have the capability to handle the maximum PWS leakage that would be expected following an earthquake. Sufficient power and a water source in the event of flow by-pass will also be required. This restart issue has been implemented for K-Reactor.

DB 2.6 - Seismically Qualified Makeup System (MRS) -- Provide seismically qualified makeup for small leakages from primary system. The seismic issue is described in Section 6.1 of this report.

Fabrication and Modification Process Control

The moderator recovery storage tank is a 1950 gallon, 304L SS tank designed for 50 psig at 100 degrees F to the design rules of ASME Section VIII. Seismic analysis has concluded that the tank nozzles and the support stresses satisfy the requirements of ASME Section III, Class 2 allowable stresses for seismic, thermal, and weight loading conditions.

The pump casing, cover, and the impeller are made of cast iron, and the shaft from Type 316SS.

The MRS pumps and supports have been seismically qualified by means of an engineering walkdown inspection, and comparison with a seismic experience data base (Reference 71).

The major process piping is made of 8-inch galvanized carbon steel from the pump discharge to the top of the loop seal and the remainder of the system is constructed of Type 304L SS. The piping was designed to the requirement of the ANSI B31.3 and seismically qualified as discussed earlier.

Pre-operational testing ensures that the system is designed and constructed to meet the functional requirements. The system pumps are designed for submerged operation, as they are located at the lowest elevation of the building.

The other critical equipment of the system are above the credible flooding level of -20 foot elevation. The MRS system is made available any time heat generating assemblies are in the reactor. A reactor shutdown is required if the MRS becomes inoperable.

Summary for MRS

From the discussion above, the staff concluded that the MRS system is well designed, seismically qualified, and the materials are properly chosen. The system is one of the latest additions to the SRS and, thus, uses ASME Code and current design specifications. WSRC has issued a report (Reference 71) in response to ROMP Restart Work Item SE-6.7F, which summarizes the status of the

structural integrity of the Moderator Recovery System. This report is currently being reviewed by the DOE staff and is an open item.

Supplementary Safety System (SSS)

The Supplementary Safety System (SSS) is a backup shutdown system for the SRS reactors. The SSS can be actuated either manually or by the Automatic Backup Shutdown-Gang Temperature Monitor (ABS-GTM) or the ABS-Safety Computer. When actuated, the SSS injects a solution of gadolinium nitrate dissolved in heavy water into the reactor moderator through six spargers near the center of the reactor. Moderator circulation and diffusion distribute the nuclear poison through the reactor core.

Engineering Analysis and Design

The poison is stored in two separate pressurized tanks, each of which supplies three injection lines. Each tank is equipped with three parallel discharge valves (two explosive and one pneumatic). The poison is forced into the reactor by gas pressure maintained in each storage tank. Either half of the SSS contains sufficient poison to shut down the reactor from full power. Injection persists long enough for the poisoned moderator to circulate through the external system and reenter the core from the plenum. If spuriously activated, the gadolinium nitrate can be removed from the moderator by deionization, and the reactor can be started within several days after activation. The SSS can be actuated manually from either the reactor control console or from the two other panels in the central control room. The implementation of Charlotte Criteria Item DB-2.1 provided seismic capability to the SSS. The seismic qualification of the SSS components is described in the technical report on the structural integrity assessment of the system (Reference 72).

Fabrication and Modifications Process Control

LOCA analyses are not included in the design bases of the SSS. The probability of a simultaneous piping break and failure of the control and the safety rods to scram is too small to warrant such consideration. But it is known that, due to corrosion-related leakage problems, one train of the SSS in the L-reactor had lost its capability of ink injection, and this condition was unattended for a long time period. This problem has been corrected by extensive inspection of the SSS piping of all three reactors and replacement of the piping where defects were detected. The original piping material was of Type 304 SS, the current replacement piping is Type 304L SS, which is known to be less susceptible to IGSCC.

The components of the SSS systems are seismic Class I, the ink storage tanks are ANS safety Class 2, and the principal construction code is ASME Section VIII-1983, with no code stamp. The nitrogen tanks are seismic Class I, ANS Safety Class 3; the construction code is unknown. The piping is seismic Class I, ANS safety Class 2; the code is ASA B31.1 or ANSI B 31.3, depending on the date of installation (Reference 72).

Liquid (Dye) Chemistry

The dye that was used until recently, Rhodamine B dye, was found to contain high levels of chlorides. The chloride content was attributed to pitting detected in the L-Reactor probe pots. The use of this dye was discontinued in 1989. The substitute dye has a very low level of chloride content that is within the acceptance standards. This action is expected to reduce the rate of corrosion significantly. Augmenting the above change, additional pH and impurity controls, QA limits, and improved mixing techniques have been introduced as upgrade measures.

Weld Examinations

During the period between March 1989 and July 1989, the SSS injection lines and probe pots weldments received thorough examinations (RT, PT and UT) as a result of discovery of a rust-through condition of the probe pots of the far side train in the L-Reactor.

The L-Reactor injection lines and pots received the most thorough examination: 121 welds were examined out of a total of 127 (95 percent). In addition to the probe pot defects described above, other localized pitting of weld material, and poor workmanship in general were recorded. As a result three injection lines in the pin room were replaced with type 304L SS piping.

In K-Reactor, 51 out of 142 welds (37 percent) were examined. Eighteen welds (13 percent) were examined in P-Reactor. Minor weld defects and some inside diameter scoring were detected, but no corrosion was evident to warrant any pipe replacement. The reduction in the level of examination at K- and P-Reactor piping was primarily due to lack of finding of any significant defect in the sampling procedure.

Testing/Inspections

Inspections and tests are performed to a varying degree of frequency according to established DPSOLs (1851A, 2608, 1106 and others) and range from daily to 4-year intervals. In addition special checks, walkdowns, and replacements are performed to established procedures to ensure the integrity of the systems.

Allegations on Corrosion Evaluations

On September 12, 1990 representatives of the DOE Office of Nuclear Safety interviewed two pipe fitters employed by MK-Ferguson, a WSRC subcontractor. These individuals alleged that stainless steel piping was installed in the Supplementary Safety System (SSS) without the required corrosion evaluations. A DOE review of this issue determined that there was some new piping in the SSS for which corrosion evaluations had not been completed and that WSRC was aware of this. WSRC has evaluated the lack of corrosion evaluations and determined that the piping is satisfactory "as is" for at least one fuel cycle. Following the first fuel cycle, either the corrosion evaluations will be completed or the piping will be replaced. The WSRC position is being documented in a nonconformance report (NCR) justifying operation with this

pipng "as is" for one cycle of operation. DOE will review the disposition of this NCR before restart. This is an open item.

Summary for SSS

Through its review, the staff concluded that the SSS has undergone substantial redesign and changes. Liquid chemistry, specifically chloride content, has been a source of corrosion and is going through an upgrade program. WSRC has issued a report (Reference 72) in response to ROMP Restart Work Item SE-6.7F, which summarizes the status of the structural integrity of the Supplementary Safety System. This report is being reviewed by the DOE staff and is an open item. In addition, as discussed above, the evaluation of the disposition of the NCR on incomplete corrosion evaluation is an open item.

6.7.3.3 Restart Criterion 3 -- Cooling Water System (CWS)

This restart criteria requires demonstration that structural integrity of the CWS is maintained under normal and abnormal load conditions using applicable codes, standards and analysis methods, including those specified in the seismic section of the SER. Components which are part of the CWS shall be subject to (1) periodic inspection and testing of areas important to safety to assess their leak tight integrity, and (2) periodic evaluation based on surveillance of the condition of the material.

The CWS supplies cooling water to the process water pump motors and diesel generators and to miscellaneous auxiliary systems and equipment in addition to its main function of cooling the PWS heat exchangers. Water from the Savannah River is supplied to the reactor areas to cool the moderator in the main heat exchangers and for other area uses. River water is pumped into the area reservoir and pumped out of the reservoir to the reactor building. Returning cooling water overflows a weir in the effluent sump and then flows back to the river via effluent streams and swamps or to a cooling pond. Backup storage of cooling water and service water for each reactor area is provided by the area reservoir (186 basin) with a capacity of 25 million gallons.

Engineering Analysis and Design

The CWS shares several components with the ECS. The cooling water basin withstands normal operational and natural phenomena, like earthquake, tornados, wind, and local site effects that could potentially disrupt its function. A complete description of the CWS can be found in the SAR.

The CWS has features to prevent freezing of the moderator in the PWS by a recirculation mode for a portion of its water. The elevation of the cooling water basin permits sufficient gravity flow through the heat exchangers when the level is two thirds full. The cooling water pressure in the process heat exchanger is lower than the PWS pressure to prevent the in-leakage of river water into the moderator and the resulting degradation of the moderator, if the heat exchanger tubes leak. The following systems are supplied with cooling water:

- o Process Water System (PWS)

- o Electrical Distribution System (Emergency Diesel, GM, Generators)
- o Water Disposal System (sump pump shaft bearing cooling)
- o Shield System (top, bottom, and thermal shields of the reactors)
- o Fuel Handling System (disassembly basin pumps and HXs),
- o Process Room Spray System (spray pump suction source)
- o Instrument Air System (air compressor cooling)
- o Emergency Cooling System (emergency and the booster pumps suction source)

The following restart issues have been identified to be applicable to the CWS.

DB 2.7 - Seismically Qualified Cooling Water System (CWS) -- Ensure that the shutdown cooling functions of the cooling water system is seismically qualified by analyses, and by DOE approved walkdowns etc. Seismic issues are described in Section 6.1 of this report.

ROMP Restart Work Item SCN No. 0214 - CWS Fracture Mechanics - - RIMP Issue No. 780 recommends that fracture mechanics analysis be done for the CWS and the ECS piping material considering an operating temperature of 40° F for the circulating water. Evaluation of the fracture toughness of the carbon steel portions of the CWS and ECS is required to examine the potential for brittle fracture of the piping. WSRC report DPST-88-353 (Reference 73) was identified as the deliverable addressing this issue. WSRC has subsequently issued a technical report (Reference 69) that supersedes Reference 53 and provides the basis for the structural integrity of the CWS piping. As stated previously, this report is being reviewed by the staff.

Fabrication and Modification Process Control

The following are the important fabrication features of the CWS.

- o Each group of cooling water pumps discharges into pipes buried deep into the ground to reduce the possibility of damage from such things as construction activities.
- o The small heat exchanger effluent lines, cooling water effluent headers, and the effluent recirculation line have bellow-type expansion joints.
- o The cooling water pipe is made of carbon steel. A small part of the large line from the PWS HXs is coated inside with Neoprene to prevent corrosion by hot water. The recirculation pipe from the effluent sump to the cooling water basin is made of carbon steel. Corrosion/erosion allowance has been made for the wall thickness of the uncoated portion of the CWS piping.

Water Chemistry

The water in the pump suction well is chlorinated to control biofouling by the injection of sodium hypochlorite from a 10,000 gallon tank. Four vertical diffusers in each pump suction well distribute the sodium hypochlorite. RSIP item B151 (Reference 74) includes provision for adding strainer pressure instrumentation in the cooling water booster pump building, strainer bypass lines, and other improvements that will allow better operator monitoring, controls, and shutdown of pumps if the pressure limits are exceeded.

In-Service Inspection (ISI) for CWS

The following items are system-integrity-related ISI details for CWS.

- o The system design and layout permit access to major system components and piping in the reactor building during shutdown.
- o Periodic tests and inspections for the essential CWS equipment are performed to written procedures.
- o Pipe supports and hangers are inspected annually to preclude any brittle fracture of the large cooling water pipes.
- o Ultrasonic thickness testing of the larger pipes is performed every 3 years.
- o The effluent recirculation pump casing and discharge column thickness is UT checked every 5 years.
- o The bellows joints are dye-penetrant tested every 3 years.

The following restart items (WISR) augment the ISIs stated above.

WISR No. 6 - Pump Degradation -- The ECS, CWS, and the Sump Pumps should be monitored and/or tested periodically for flow degradation. This item was completed and is covered in SER Section 14.0, Inspection and Testing.

WISR No. 38 - LOPA Event Frequency -- Institute a visual inspection and surveillance program to ensure the integrity of the CWS piping. The piping of concern is that which, if it failed, could lead to process water pump flooding and subsequent failure to provide circulation. A CWS piping inspection and surveillance program satisfying the WISR-38 requirements was implemented for the K-Reactor (see WSRC RTM-5001, Rev. 1, "Cooling Water System Inspection Review," June 25, 1990). This item is discussed in SER Section 14.0, Inspection and Testing.

Summary for CWS

The staff concluded that the reliability of the CWS is acceptable, pending resolutions of the following outstanding issues.

- o Fracture toughness of the carbon steel material of the piping is a concern that is being addressed by WSRC through RIMP No. 780. WSRC has issued a report (Reference 69) in response to ROMP Restart Work Item SCN No. 0214, which provides a fracture mechanics assessment of the carbon steel piping in both the Cooling Water System and the Emergency Cooling System. This report is being reviewed by the DOE staff for approval, as previously stated.
- o The CWS pump flow degradation is a concern that is being evaluated and resolved prior to restart through WISR-6. A pump flow monitoring program was implemented as required by WISR-6 (Reference WSRC document RTM-5005, "WISR-6 Pump Degradation K-Area," dated September 15, 1989). Refer to SER Section 14.0 for the evaluation of this topic.
- o The visual inspection and surveillance program of the CWS piping being implemented through WISR-38 shall ensure the integrity of the piping, reducing the chance of a LOPA due to flooding the pump rooms as a result of piping failure. As stated previously, a surveillance program for the CWS piping has been established. DOE staff review of this program, as reported in SER Section 14.0, has concluded that it satisfies the restart criteria for inservice inspection and testing of CWS piping.

WSRC has issued a report (Reference 75), in response to ROMP Restart Work Item SE-6.7G, which summarizes the status of the structural integrity of the Cooling Water System. This report is being reviewed by the DOE staff and is an open item.

6.7.4 Conclusion

In the process of conducting this evaluation, the staff has been assisted by a panel (DOE Independent Review Committee [DIRC] for System Integrity) of nationally recognized experts in the areas of metallurgy, fracture mechanics, corrosion, and nondestructive examination (NDE). The review panel consisted of the following members:

- o Dr. Paul Shewmon, Ohio State University (Chairman of DIRC)
- o Dr. Steve Doctor, Pacific Northwest Laboratories
- o Dr. Gery Wilkowski, Battelle Columbus Laboratories
- o Dr. Akram Zahoor, Novetech Corporation

The Panel conducted an extensive and thorough review of the WSRC system integrity programs. A final report, issued on July 26, 1990, provided a positive endorsement of the WSRC methodology used to demonstrate system structural integrity.

The major conclusions drawn from the staff review, including the conclusions from the DIRC investigation, were as follows:

- o The documentation of the structural integrity of the SRS reactor systems and the analysis therein provides a sound basis for confirmation of the integrity of the tanks, piping, and components during normal and anticipated abnormal conditions.
- o Given the design, materials of construction, leak detection, and in-service inspection (ISI) program, the sudden (instantaneous) formation of large breaks in the pressure boundary is so unlikely as to be almost incredible, for any foreseen operating conditions.
- o A very conservative program is being followed to set up the ISI plan.
- o The tritium leak detection system is very sensitive and is able to detect extremely small leaks. The tritium leak detection system will allow sufficient time for a safe reactor shutdown in case of abnormal leakage from the primary system.

Based on the above evaluation, the staff concludes that adequate progress is being made to address this issue. Resolution of the open items, identified above, will be documented in a supplement to this SER.

6.7.5 References

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2. WSRC, WSRC-RP-90-559, Topical Report: "Structural Integrity of the Process Water System (PWS) of the Savannah River Site Production Reactors (U)," June 30, 1990.
3. WSRC, DPST-87-852 (ECS-LOCA-2), "System Description."
4. WSRC, DPST-87-837 (ECS-LOCA-3), "LBB Large Piping."
5. WSRC, DPST-87-822 (ECS-LOCA-4), "Water Hammer Analysis."
6. WSRC, DPST-87-886 (ECS-LOCA-5), "Thermal/Mechanical Fatigue."
7. WSRC, DPST-87-859 (ECS-LOCA-6), "Sensitization and Ferrite Measurement."
8. WSRC, DPST-88-404 (ECS-LOCA-7), "LBB, Flanges, Pumps, and Valves."
9. WSRC, DPST-88-472 (ECS-LOCA-8), "LBB Heat Exchangers."
10. WSRC, DPST-88-434 (ECS-LOCA-9), "LBB Expansion Joints."
11. WSRC, DPST-88-455 (ECS-LOCA-10), "LBB Tank Upper Plenum."
12. WSRC, DPST-88-546 (ECS-LOCA-11), "Indirect Pipe Break Probability (K)."
13. WSRC, DPST-88-468 (ECS-LOCA-12), "Direct Pipe Break Probability."

14. WSRC, DPST-88-400 (ECS-LOCA-13), "Indirect Pipe Break Probability (for L- and P-Rxs)."
15. WSRC, DPST-88-328 (ECS-LOCA-14), "Pipe Break Dynamics."
16. WSRC, DPST-88-462 (ECS-LOCA-15), "Alternate LOCA."
17. WSRC, EED870191 (ECS-LOCA-16), "Present Inspection Procedures."
18. WSRC, DPST-88-675 (ECS-LOCA-17), "Recommended Inspection Guidelines."
19. WSRC, DPST-87-895 (ECS-LOCA-18), "Heat Exchanger C-Clamp Testing."
20. WSRC, DPST-88-445 (ECS-LOCA-19), "System Leak Rate Significance."
21. WSRC, DPST-88-212 (ECS-LOCA-20), "Primary Coolant System Analysis."
22. WSRC, DPST-88-825 (ECS-LOCA-21), "Water Hammer Analysis- Phase II Results."
23. WSRC, DPST-88-329 (ECS-LOCA-22), "As-Built Verification."
24. WSRC, DPST-88-426 (ECS-LOCA-23), "Leak Detection System."
25. WSRC, DPST-88-479 (ECS-LOCA-24), "P-Reactor Startup Vibration Measurement."
26. WSRC, DPST-88-855 (ECS-LOCA-25), "Heat Sink Welding."
27. WSRC, DPST-88-377 (ECS-LOCA-26), "LBB Small Piping."
28. WSRC, DPST-88-336 (ECS-LOCA-27), "Small Piping Stresses."
29. WSRC, DPST-88-469 (ECS-LOCA-28), "Direct Component Failure Probability."
30. WSRC, DPST-88-394 (ECS-LOCA-29), "ECS Piping Stresses."
31. WSRC, DPST-86-718 (ECS-LOCA-30), "P-, K-, L-Tank Integrity."
32. WSRC, DPST-86-747 (ECS-LOCA-31), "L-Tank Fracture Analysis."
33. WSRC, DPST-88-385 (ECS-LOCA-32), "Calculated Flange Moments."
34. WSRC, DPST-88-414 (ECS-LOCA-33), "LBB Large Pipes with Large Leaks."
35. WSRC, DPST-88-571 (ECS-LOCA-34), "Large Pipe Castings."
36. WSRC, DPST-88-435 (ECS-LOCA-35), "Embedded Piping Leak Areas."
37. WSRC, DPST-88-264 (ECS-LOCA-36), "Water Hammer Analysis- Phase II Plan."
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40. WSRC/SRL, SRL-EDG-90-290, Task 90-019-1, Subtask 2, "Summary of Engineering Analysis and Design for the Process Water System," September 17, 1990.
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43. WSRC, WSRC-RP-89-236, "Topical Report Leak Before Break Capabilities and Related Safety Implications for the Savannah River Production Reactors (U)," May 1989.
44. WSRC, WSRC-RP-89-152, "K-Reactor Tank Integrity Status (U)," Rev. 1, May 1989.
45. WSRC, DPST-89-356, "Process Water System Integrity - A Metallurgical Evaluation," March 1989.
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47. WSRC, DPST-97-469, "Incidence of IGSCC In Reactor Tanks and Piping," December 1987.
48. WSRC, "Savannah River Site Production Reactor Safety Analysis Report," July 17, 1989.
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53. WSRC, WSRC-RP-90-91, "Reactor Materials Program Leak-Before-Break Analysis of the Process Water System Piping," March, 1990.
54. WSRC, OPS-RES-90-0141, "Methods, Systems, and Procedures for Moderator Chemistry Control at SRS," May 18, 1990.

55. WSRC, OPS-ROD-890246, "Moderator Chemistry Surveillance and Analysis Requirements," December 27, 1989.
56. WSRC, OPS-RES-900372, "Assessment of Methods, Systems, and Procedures for Moderator Chemistry Control at SRS," August 10, 1990.
57. WSRC, OPS-ROD-900073, "Assessment of Moderator Chemistry Surveillance and Analysis Requirements at SRS," February 21, 1990.
58. WSRC, WSRC-RP-89-368, Volume 2, Revision 4, "Reactor Operations Management Plan (ROMP)," July 13, 1990.
59. WSRC, WSRC-RP-90-99, "Leak Detection System Reliability," March 9, 1990.
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61. WSRC, DPST-88-676(ECS-LOCA-1A).
62. U. S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan 3.6.1., Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," Rev. 1, July 1981.
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71. WSRC, WSRC-TR-90-321, "Structural Integrity Assessment of the Moderator Recovery System K-, P-, and L-Reactors - Savannah River," Sept., 1990.

72. WSRC, WSRC-TR-90-322, "Structural Integrity Assessment of the Supplementary Safety System - K-, P-, and L-Reactors - Savannah River Site," September 1990.
73. WSRC, DPST-88-353, "Fracture of 100 Area Carbon Steel Piping - A Lower Bound Analysis."
74. WSRC, WSRC-RP-89-379, "Development and Management of Reactor Safety Improvement Program (RSIP)," July 1989.
75. WSRC, WSRC-TR-90-324, "Structural Integrity Assessment of the Cooling Water System - K-, P-, and L-Reactors - Savannah River Site," September 1990.
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7 FIRE PROTECTION

7.1 Background

The Savannah River Site (SRS) reactor building is essentially an open interior, noncombustible structure with limited fixed combustible loadings. Principal combustibles present are lubricating oils, fuel oils, cables, and transient combustibles. Because of the confinement system configuration, the building is subdivided into three open areas, where effluent is drawn from lower to higher radiological zones. There are few totally enclosed areas to passively limit fire propagation.

Due to the historical emphasis on fire prevention versus fire protection, few automatic fire suppression and detection systems are present in the reactor building. There are several Halon suppression systems for the diesel buildings and select computer rooms. Manual fire suppression for the reactor building is provided by way of portable fire extinguishers and limited capacity water hoses. The manual suppression capability and other measures are utilized by a combined response of area fire support teams and the Savannah River Site Fire Department.

Fire protection activities required by the reactor restart criteria include; (1) installing a dry riser in one stairwell, (2) installing a fire water supply tap into an existing service water pipe, (3) installing heat detection capability in select high radiation areas, (4) establishing fire patrols and watches, (5) verifying that existing fire protection systems are functional, and (6) providing procedures and administrative control upgrades.

In addition, a limited scope post-fire safe shutdown analysis that addressed the Title 10 of the Code of Federal Regulations, Part 50 (10CFR50), Appendix R, Section III G, III J, and III L (Reference 1) was performed.

7.2 Restart Criteria

The criteria for determining the adequacy of the SRS fire protection system for restart are described below.

- (1) All Savannah River K-Reactor Restart Strategy (Reference 2), referred to herein as the Charlotte Criteria items for fire protection, shall be resolved.
- (2) A safe shutdown analysis shall be prepared for each operating area to identify and qualify the postulated limiting-worst case set of fire events. The safe shutdown analysis results and dispositioning of recommendations shall ensure that no postulated single fire event can compromise the reactor facility such that:
 - (a) The ability to achieve and maintain reactor safe shutdown conditions is in doubt.
 - (b) Reactor operations personnel can be prevented from performing all actions required to achieve and maintain reactor safe shutdown.

This analysis, which is performed to support reactor restart, shall consider postulated fire events when off-site power is available as well as when there is a coincident loss-of-off-site power. (See SER section 6.3.1, "Electrical Power Systems," for the definition of on-site and off-site power.)

- (3) The abnormal condition control procedures, including corrective action performance time sequences (as appropriate), shall be prepared to address the limiting-worse case shutdown scenarios.
- (4) Life safety upgrades shall be provided through egress marking and training, roving fire patrols, and installing portable fire extinguishers.
- (5) An on-site auxiliary fire brigade shall be trained, equipped, and staffed as defined by the approved Safety Analysis Report (Reference 3).
- (6) Reactor area operating, emergency, surveillance, maintenance and testing procedures shall be developed and implemented for all fire protection systems and equipment implemented as part of the reactor restart program.
- (7) Training programs for the reactor area and, as applicable, site-wide personnel, shall be developed and implemented for all fire protection systems and procedural programs implemented as part of the reactor restart program.
- (8) Westinghouse Savannah River Company (WSRC) shall establish a strict control program for transient combustibles. That program shall be implemented for the following areas (as a minimum) by restart: reactor areas 105, 108, 151, and 190. The program shall require periodic inspection by WSRC for compliance with the transient combustible control program.
- (9) A facility fire protection readiness assessment by a certified fire protection engineer from an organization independent from WSRC shall be performed approximately one month before restart. This assessment shall ensure compliance with reactor restart fire protection commitments.

7.3 Evaluation

7.3.1 Restart Criteria Program Evaluation

Restart Criteria 1 -- Charlotte Criteria

The following evaluates the programmatic aspects of items prescribed by the Charlotte Criteria DB-4.1 to 4.10 which is scoped and scheduled by the WSRC Reactor Operations Management Plan (ROMP), WSRC-RP-89-368 (Reference 4).

(1) DB-4.1 -- Periodic Inspection for Combustibles

The staff reviewed WSRC procedure number RDP-1.11, "Reactor Areas Fire Protection Program Plan" (Reference 5), and DPSOL 100-4603-PLK, "Fire Protection Inspections" (Reference 6), and determined that the requirements and guidelines for periodic inspection of combustibles identified within the program plan and procedures addresses the requirements of Charlotte Criteria DB-4.1. Pending formal issuance of RDP-1.11, this issue remains open.

(2) DB-4.2 -- Pre-Startup Certified Fire Control Engineer Walkdown

The staff reviewed WSRC Fire Protection Oversight memorandum ESH-FPC-89-2022, "K-Reactor Fire Inspection Report and SFPE Membership," dated August 2, 1989 (Reference 7), and determined that the report partially satisfies the requirements of Charlotte Criteria DB-4.2 for combustible walkdowns. The WSRC provision of guidelines and criteria utilized to perform the walkdown remains an open item.

The staff also reviewed WSRC's, "Fire Protection System Readiness Report," ESH-FPO-90-0615, Revision 2, dated October 26, 1990 (Reference 8), and determined that the report partially satisfies the requirements of Charlotte Criteria DB-4.2 for system readiness. The staff reviewed WSRC-RP-89-368, (ROMP) DB-4.2, Deliverable 1, Revision 5, dated October 5, 1990, and determined that the defined fire protection engineers qualifications and the method by which all members of this effort will be identified as certified fire control engineers satisfies the requirements of Charlotte Criteria DB-4.2 for conducting walkdowns by certified fire control engineers. The WSRC provision of guidelines and criteria utilized to perform the walkdown remains an open item.

(3) DB-4.3 -- Maintenance/Construction Fire Watch Procedures

The Charlotte Criteria requires that a review of procedures be conducted and documented to require fire watches for maintenance and construction work using ignition sources. WSRC has conducted the review and has committed to revise and develop a new procedure to conform with NFPA 51B, "Standard for Fire Prevention in Use of Cutting and Welding Processes" (Reference 9) requirements. Pending issuance of a procedure which conforms with NFPA 51B, this remains an open item.

(4) DB-4.4 -- Fire Equipment Surveillance Procedures

The staff reviewed DPSOL 100-1879C-K, "Monthly Patrols" (Reference 10), DPSOP 196, "Fire Protection Division Operating Procedures" (Reference 11), and DPSOP 196-1, "Fire Protection DPSOLs" (Reference 12), which describes fire protection. The staff determined that these procedures partially satisfy the requirements of Charlotte Criteria DB-4.4. The provision of completed surveillance procedures for existing and newly installed fire detection and suppression systems remains an open item.

(5) DB-4.5 -- Fire Emergency Procedures

WSRC has incorporated fire emergency procedures into DPSOP 67 (Reference 13) "100 Areas Emergency and Disaster Plans, DPSOLs 100-4000 through 4450," and other site emergency manuals. Staff review of this document verified that this satisfies the requirement of DB-4.5.

(6) DB-4.6 -- Control of Introduction of Combustibles

The staff reviewed DPSOL 100-4602-PLK, "Control of Transient Combustibles," Revision 3 (Reference 14). Based on this review, the staff determined that the guidelines and requirements for control of introduction of combustibles identified in the procedure satisfy the requirements of Charlotte Criterion DB-4.6.

(7) DB-4.7 -- Combustible Scaffolding

Charlotte Criteria DB-4.7 requires that all combustible scaffolding, including scaffolding painted with fire retardant paint be removed from the reactors. WSRC provision of program guidelines that will ensure the removal of all combustible scaffolding from the 105, 108, 190 and 151 buildings prior to restart remains an open item.

(8) DB-4.8 -- Review of Fire Detection and Suppression Equipment and Fire Fighting Procedures

The staff has reviewed the independent critical review document commissioned by WSRC, which concluded that a dry riser system would be provided in one stairwell and fire detection would be installed in specific areas inaccessible during normal reactor operations. In general, where detection systems are not installed, all operating accessible building areas would rely on roving fire patrols. This WSRC review partially satisfies Charlotte Criteria DB-4.8. The requirement that this WSRC review be performed by a certified fire protection engineer is satisfied by WSRC-RP-89-368, which defines the qualifications of a fire protection engineer and the methods by which members are identified as certified fire protection engineers.

The provision of a WSRC review document relative to firefighting procedures to assure the adequacy of training, equipment, and procedures remains an open item.

(9) DB-4.9 -- Review by Certified Fire Protection Engineer to Determine Adequate Fire Detection in Reactor Buildings

The provision of guidelines and criteria to be utilized in performance of a WSRC review to document the assurance that sufficient fire detection to support SRS program objectives is in the reactor building remains an open item.

(10) DB-4.10 -- Shutdown on Confirmed Fire Alarm

The staff has reviewed WSRC procedure DPSOL 100-ME-660-PLK, "Prompt Response to Reactor Facility Fire" (Reference 15), which requires prompt response to any indication of a fire and requires the reactor to be shutdown if a fire is confirmed, indeterminate, or the alarm site is inaccessible. The staff concluded that this procedure satisfies the requirements of Charlotte Criteria DB-4.10.

Safe Shutdown Analysis (SSA)

WSRC hired a subcontractor with substantial experience in conducting and reviewing commercial nuclear industry Title 10 of the Code of Federal Regulations, Part 50, (10CFR50) Appendix R, studies. This individual was tasked with organizing and directing the WSRC SSA effort under the cognizance and direction of Reactor Engineering. The staff determined that this individual's knowledge and experience was sufficient for an Appendix R review, which is a first-of-a-kind for the reactors at SRS.

The SSA is an Appendix R review modified to account for the SRS production reactor dissimilarities compared to light water reactors for which Appendix R was written. A limited scope post-fire safe shutdown analysis that addressed 10CFR50, Appendix R, Sections III G, III J, and III L was performed. The approach taken was to identify those aspects of Appendix R that had no direct relationship to the SRS production reactors, discuss the differences, and establish a method that complies with the intent of Appendix R. This analysis resulted in the addition of the refueling equipment (charge and discharge machines) to the SSA. This system is substantially dissimilar to refueling equipment for light water reactors (LWRs). The staff reviewed the methodology used by WSRC and agreed that the approach was valid and acceptable with one noted exception. The staff has recommended that WSRC consider lube oil collection requirements, because reference procedures (DPSOL 105-1141-PLK) cause operators to initiate refilling without investigation and the low-level alarm setpoint is 95 gallons. This is an open item.

After the basic methodology was set, WSRC began to establish a bounding set of equipment necessary for safe shutdown given any single credible fire. Power, instrument, and control cables for the safe shutdown equipment were identified; associated circuits and cabling were also identified. Fire areas for the safe shutdown analysis were defined. Each fire area was analyzed for postulated credible fire with and without reactor area power. This analysis included the ability of reactor operators to perform necessary safe shutdown plant manipulations given the fire.

Potential vulnerabilities to safe shutdown capability were identified for further analysis. The potential vulnerabilities not only included direct fire damage to safe shutdown equipment, but also included loss of safe shutdown capability due to cable damage resulting in the loss of function and spurious or erratic operation of safe shutdown components not directly damaged by the fire. Based on a detailed analysis, WSRC determined whether the condition was acceptable as-is (did not prevent safe shutdown) or whether modifications to

the plant hardware or the supporting procedures were necessary to eliminate the risk. Figure 2.7-1 in WSRC-RP-90-195, "Fire Protection - Safe Shutdown Analysis for K-Reactor" (Reference 16), illustrates the review process.

Upon completion of the analysis, WSRC established a process to review the SSA, the proposed plant modifications, and the disposition of modifications (pre-restart/post-restart). Included was the long-term maintenance (document control, design control) of SSA integrity. Figure 1.4-1 to WSRC-RP-90-195 illustrates the review/approval/dispositioning process. This issue remains open pending completion of the SSA. Additionally, the following are open items:

- o WSRC has not yet documented the programs and procedures that maintain the SSA as a controlled document and assure that future design changes to the plant that may affect the SSA are captured and appropriately dispositioned so that the SSA assumptions and conclusions remain valid.
- o WSRC has not yet documented the program to justify continued reactor operations with the existing vulnerabilities identified in the fire protection systems. The program was contained within the RIMP process illustrated in the Safe Shutdown Analysis Figure 1.4-1 to WSRC-RP-90-195.
- o WSRC has not yet documented the program that implements the procedural, administrative and system changes identified in the Safe Shutdown Analysis.

Abnormal Condition Control Procedures

The WSRC provision of a fully developed set of Abnormal Condition Control Procedures remains an open item.

Life Safety Upgrades

The WSRC provision of a plan for life safety upgrades to be implemented prior to reactor restart remains an open item.

On-Site Auxiliary Fire Brigade

Sections 9.5.1.5 and 13.3.3 of the Safety Analysis Report (SAR) describe the function and utilization of On-Site Auxiliary Fire Brigades (also called Fire Support Teams). The staff reviewed the following documents in order to assess whether the WSRC program for the Auxiliary Fire Brigades was sufficient to maintain those duties as described in the SAR:

- o DPSOP 67-1, "Fire Control Plan: 100 Areas" (Reference 17)
- o DPSOP 196, "Fire Protection Division Operating Procedures" (Reference 11)

- o DPSOP 178, "Fire Control Plan for Savannah River Plant" (Reference 18)
- o Savannah River Site Fire Department Operations (Reference 19)

Per the staff review, the procedures define the SRS fully trained and equipped fire department credited in the SAR. The procedures define training levels of fire support teams for the K-Reactor (100 areas), but do not define the auxiliary fire brigades credited in the SAR. This area will remain open pending formal issuance of SRS Fire Department Operations. Additionally, the following open items remain:

- o Information to define and assess the responsibilities, qualifications, training and staffing requirements for the on-site auxiliary fire brigade.
- o Information to assess equipment requirements for the on-site auxiliary fire brigade.
- o Information to assess how and when the on-site auxiliary fire brigade is invoked.
- o Information to assess how and when the SRS fire Department is invoked.
- o Information to assess the integration and interactions of the on-site auxiliary fire brigade, the SRS Fire Department, and the Reactor Operations Department.

Reactor Restart Systems and Equipment Procedure Upgrades

The following restart criteria require procedural upgrades due to changes in plant configuration:

- o Restart Criterion 1 -- Charlotte Criteria DB-4.2, DB-4.4, DB-4.6, DB-4.8, and DB-4.9
- o Restart Criterion 3 -- Safe Shutdown Analysis
- o Restart Criterion 4 -- Life Safety Upgrades
- o Restart Criterion 5 -- On-site Auxiliary Fire Brigades

The staff has not yet received sufficient information from WSRC to fully assure that necessary procedural upgrades will be completed for reactor restart fire protection systems and equipment. This will be an open item.

Training Programs

The following restart criteria require the development of training programs due to changes in plant configuration:

- o Restart Criterion 1 -- Charlotte Criteria DB-4.1, DB-4.3, DB-4.4, DB-4.6, DB-4.8, DB-4.9 and DB-4.10
- o Restart Criterion 3 -- Safe Shutdown Analysis
- o Restart Criterion 4 -- Life Safety Upgrades
- o Restart Criterion 5 -- On-site Auxiliary Fire Brigade
- o Restart Criterion 6 -- Reactor Area Emergency, Operating, Surveillance, Maintenance and Testing Procedures
- o Restart Criterion 8 -- Program for Control of Transient Combustibles and Periodic Inspections for Combustibles

The staff has not received sufficient information from WSRC to assure that necessary training will be conducted prior to restart. This is an open item.

Transient Combustible Control Program

The staff reviewed DPSOL 4602, "Control of Transient Combustibles," Revision 3, and DPSOL 4603, "Fire Protection Inspections," and determined that the guidelines and requirements for a program to control the introduction of transient combustibles and the periodic inspection for combustibles are sufficient to satisfy the requirements of Restart Criteria 8.

Fire Protection Readiness Assessment

WSRC provision of programmatic requirements and methodology of the readiness assessment, reviewer criteria, and a resolution process for any findings that may be generated will be tracked as an open item.

7.4 Conclusion

The completion and approval of the safe shutdown analysis and adequate dispositioning of its recommendations assures that a fire will not create an unacceptable risk to the public health and welfare upon reactor restart. Implemented procedural upgrades will help minimize the risk of fire by strictly controlling ignition sources and transient combustibles. The addition of linear thermal detectors, a dry riser, additional portable fire extinguishers, roving fire patrols, fire watches when required, additional training, and operating procedural upgrades will provide improvements in the detection, mitigation and response to fire. Remaining open items are clearly identified and their closure will be addressed in a supplement to the SER.

7.5 References

1. Title 10 Code of Federal Regulations, Part 50, Section 48, Appendix A and Appendix R.
2. WSRC, K-Reactor Restart Strategy (Charlotte Criteria), November 25, 1988.

3. "SRS Production Reactor Safety Analysis Report."
4. WSRC, WSRC-RP-89-368, "Reactor Operations Management Plan," November 5, 1990.
5. WSRC, Procedure RDP 1.11, "Reactor Areas Fire Protection Program Plan" (Draft 0-B).
6. WSRC, DPSOL 100-4603-PLK, "Fire Protection Inspections."
7. WSRC Memorandum, ESH-FPC-89-2022, "K-Reactor Fire Inspection Report and SFPE Membership," August 2, 1989.
8. WSRC, ESH-FPO-90-0615, "Fire Protection Readiness Report," Revision 2, October 26, 1990.
9. National Fire Protection Association, NFPA 51B-1984, "Standard for Fire Prevention in Use of Cutting and Welding Processes."
10. WSRC, DPSOL 105-1879C-K, "Monthly Patrol" Rev. 0, April 27, 1990.
11. WSRC, DPSOP 196, "Fire Protection Division Operating Procedures," Rev. 25, April 1989.
12. WSRC, DPSOP 196-1, "Fire Protection Division DPSOLs," Rev. 0, January 1984.
13. WSRC, DPSOP 67, "100-Areas Emergency and Disaster Plans, DPSOLs 100-4000 through 4450."
14. WSRC, DPSOL 100-4602-PLK, "Control of Transient Combustibles," Rev. 3, April 12, 1990.
15. WSRC, DPSOL 105-ME-660-PLK, "Prompt Response to Reactor Facility Fire," Rev. 1, September 15, 1989.
16. WSRC, WSRC-RP-90-195-DRAFT, "Fire Protection - Safe Shutdown Analysis for K Reactor," Rev. 1 (UCNI).
17. WSRC, DPSOP 67-1, "Fire Control Plan 100 Areas (UCNI)."
18. WSRC, DPSOP 178, "Fire Control Plan for Savannah River Plant," Rev. 6, April 1989.
19. WSRC, "Savannah River Site Fire Department Operations," undated.

8 TESTING AND STARTUP

8.1 Control of Testing

8.1.1 Background

The Savannah River Site (SRS) reactor facilities will have been shutdown for approximately 2 years by the date of restart. Considerable modifications have been conducted during the outage. Both the Department of Energy (DOE) and Westinghouse Savannah River Company (WSRC) recognize the need for significant testing prior to restart. WSRC has committed to ensuring adequate review and approval of the test requirements and procedures used to conduct testing as well as the administrative controls used to control testing in the field. DOE approval of the WSRC test authorizations (TAs) for all reactor areas is required, and all testing will be controlled by formalized requirements using previously approved operating, surveillance, special test, and startup procedures. In addition, Reactor Area Operations management is required to authorize all testing conducted within the area.

The need to conduct selected testing was defined in the Westinghouse Independent Safety Review (WISR). Specifically, the WISR (1) defined the need for loss-of-coolant-accident (LOCA) performance testing and for emergency cooling system (ECS), cooling water system (CWS), and sump pumps to be tested or to have documented results of previously conducted tests to satisfy the need for testing; (2) identified battery and uninterruptible power supply (UPS) testing to be performed in accordance with the consensus industry standard; and (3) defined more rigorous and detailed startup core testing.

To properly control the testing scope definition, all testing requirements developed to respond to WISR recommendations were evaluated by the Restart Issue Management Program (RIMP), and the results of these committee evaluations were factored into WSRC's schedule change process. High priority integrated safety evaluation program (ISEP) topics were evaluated using the same process. All testing controls required by the Charlotte Criteria were to be applied to these and all other restart tests conducted in the reactor areas.

Fundamentally, two predominant philosophical changes were addressed relative to historical SRS testing methods. Both will be incorporated into the control of restart testing for SRS reactors. The first, the draft DOE Nuclear Safety Policy DOE N 5480.PP, dated May 2, 1989 (Reference 1), described the need for facility-wide testing versus discrete component or subsystem functional tests; the second is the adoption of engineered safety feature (ESF) design-based performance testing versus component level functional testing. The adoption of these two philosophies affects not only the scope of testing performed at SRS but the need to more rigorously define test requirements, develop test procedures, control testing in the field, evaluate test results, and formally disposition out-of-tolerance or nonconforming results.

8.1.2 Restart Criteria

The criteria for determining the adequacy of the SRS program for control of testing for restart are described below.

A formal, controlled procedure shall be required which shall govern the conduct of experimental activities and tests at all reactor facilities. This procedure shall integrate, as a minimum, each of the following elements:

- o definition of the types of tests
- o test authorization (TA) and special test procedure requirements
- o requirement for DOE approval of all TAs
- o requirement that all tests be accomplished with formal procedures (special or standard operating procedures), which shall govern the operations and experimental test performance
- o requirement for DOE approval of selected special procedures
- o authorization of all testing of the reactor operations on-shift supervision
- o authorization of all experiments and special tests by the Plant Manager
- o safety review and environmental evaluation of TAs and special procedures
- o organizational responsibilities to insure proper interfacing of operations, test, and management personnel
- o requirement for qualifications and training of operations and test personnel
- o provision for appropriate hold points for safety and quality reviews
- o provision for pretest predictions where possible
- o provision for test evaluation, analysis, acceptance, and conclusions
- o documentation of all test information, data, reports, and conclusions
- o provision for post-test debriefing and evaluation of conduct of operations
- o provision for records management of all test documentation

8.1.3 Evaluation

The DOE staff reviewed SRS Control Procedure DPSOL 336, "Reactor Operations Standing Instructions" (Reference 2), for test control and Startup Test Program Manual (STPM) RD 5.2 (Reference 3), for control of testing during reactor restart to ensure that the restart criteria are enveloped by these documents. The results of the staff evaluation are as follows:

- (1) The formal control procedure for experimental activities and tests is DPSOP 336, II.12 Rev. 3, "Reactor Operation Standing Instruction No. II, 12 Test Control." A section titled "Terms and Definitions" defines various types of tests. For the Startup Test Program (STP) an administrative procedure in the Startup Test Program Manual, STPM-06, "Control of Testing," has been prepared and approved for control of restart testing and defines special tests in support of reactor restart. This meets the restart criteria.
- (2) STPM-06, "Control of Testing," addresses the requirements for test authorizations for the restart test program. However, DPSOP 336, II.12, does not address the requirements for test authorizations and special procedures or directly reference where these requirements may be found. This is an open item.
- (3) STPM-06, "Control of Testing," addresses DOE approval of all test authorizations. However, DPSOP 336, II.12, does not address the requirement for DOE approval of all test authorizations or directly reference where this requirement may be found. This is an open item.
- (4) STPM-06, "Control of Testing" requires that all tests be performed with formally prepared and approved procedures. DPSOP 336, II.12, specifically requires that all tests be performed with formal procedures. Section 2.0 of the procedure states that the specified test activities are subject to the requirements of QAP 11.1, SRP Quality Assurance Manual (Reference 4). This meets the restart criteria.
- (5) STPM-06, "Control of Testing" requires that DOE approve selected special test procedures. SRSPO test engineers are required to approve all special test procedures related to the Startup Test Program (STP) for the SRS reactors. DPSOP 336, II.12 requires DOE to approve selected test procedures. This meets the restart criteria.
- (6) DPSOP 336, II.12 requires Reactor Operations on-shift supervision to authorize all tests. STPM-06, "Control of Testing," specifies that the Reactor Operation Division (ROD) Shift Manager authorize all tests. This meets the restart criteria.

- (7) STPM-06, "Control of Testing," specifies that the Plant Manager is to release all tests for performance. DPSOP 336, II.12 requires the Plant Manager to authorize all tests. This authorization is specified in the purpose section of the procedure. (See Item 6). While this requirement is explicit, other parts of the procedure imply that test procedures may be approved indirectly by approving test schedules or by delegated individuals. Insufficient specifications are given as to how the intent of the Plant Manager's approval is maintained through the indirect or delegated authority process. This concern also exists in the Temporary Procedure Change (TPC) process. This is an open item.
- (8) DPSOP 336, II.12 and STPM-06 specify that safety review and evaluation of Test Authorizations and special procedures are required. This meets the restart criteria.
- (9) Both DPSOP 336, II.12 and STPM-06, "Control of Testing," state the responsibilities of personnel involved with testing and define the interface requirements between Reactor Startup and Testing (RSAT) and ROD personnel. This meets the restart criteria.
- (10) DPSOP 336, II.12 addresses training of personnel and some special training requirements. Although DPSOP 336, II.12 states that it covers all personnel, it refers to operations and maintenance personnel only. Training and qualification requirements for Reactor Engineering (test) personnel have not been addressed or referenced. Pretest briefings are not specified in the control procedure. This is an open item. STPM-06, "Control of Testing," states that startup personnel conducting testing shall be qualified as Test Engineers. STPM-07, "Qualification of Test Personnel," is the administration procedure that addresses training and qualifications of test personnel.
- (11) STPM-06, "Control of Testing," makes provision for quality verification points to be specified in the procedure and verified by the test engineer. DPSOP 336, II.12 provides for appropriate hold points in the test procedures for safety and quality reviews. This meets the restart criteria.
- (12) Both STPM-06, "Control of Testing," and DPSOP 336, II.12, state that discussion of pretest predictions are to be included as part of the Pretest/Shift Briefings. This meets the restart criteria.
- (13) Both DPSOP 336, II.12, "Evaluating Test Results," and STPM-06, "Control of Testing," address test results review and approval. STPM-06, "Control of Testing," clearly defines the path for test evaluation following test performance, and the responsibility for the evaluation. This meets the restart criteria.
- (14) The method of authenticating test data is specified in DPSOP 336, II.12. Requirements, such as methods of data collection and data sheet requirements, are not specified or referenced. STPM-06,

"Control of Testing," provides for proper test documentation. This meets the restart criteria.

- (15) STPM-06, "Control of Testing," specifies the need for post-test debriefing and evaluation of test operations if significant test control deficiencies are observed. DPSOP 336, II.12 addresses post-test debriefing and evaluation of conduct of operations. This meets the restart criteria.
- (16) STPM-09, "Startup Test Program Record Document Control Index," adequately describes the responsibilities and retention period for startup tests. DPSOP 336, II.12, Section 6.0, "Records," specifies the requirements for records management of test documentation. However, the responsibilities for proper records management and the path of the documentation from test completion to interim storage are not clearly defined. This is an open item.

8.1.4 Conclusion

The staff concludes that the Startup Test Program Manual meets the requirements for conduct of the startup test program and that closure of the open items specified above will be adequate to satisfy the restart criteria.

8.1.5 References

- 1. U.S. Department of Energy, "Nuclear Safety Policy", (Draft Notice), DOE N 5480.PP, May 2, 1989.
- 2. WSRC, DPSOP 336, II.12, Rev. 4, "Reactor Operations Standing Instruction No. II.12 Test Control," February 8, 1991.
- 3. WSRC, Startup Test Program Manual, RD 5.2, May 1, 1990.
- 4. WSRC, SRP Quality Assurance Manual, Section 11.1.

8.2 System/Preoperational Testing

8.2.1 Background

The Westinghouse Savannah River Corporation (WSRC), Reactor Operations Management Plan (ROMP), WSRC-RP-89-368 (Reference 1), required that a controlled, formal, and deliberate program be established for K-Reactor design testing/data control and restart/power ascension. The plan to be developed shall have the following characteristics:

- (1) identified hold points/release requirements
- (2) specifications of essential data observed and retained
- (3) specified allowable data variance and actions required for data variances exceeding limits
- (4) approval requirements for release from specified hold points
- (5) specifications of required technical evaluations
- (6) assignment of line management responsibility
- (7) QA requirements molded after national consensus standards
- (8) audit requirements to determine compliance

DOE approval of the plan before implementation is also required.

The draft DOE Nuclear Safety Policy DOE N 5480.PP (Reference 2) requires that a testing and startup program for a new nuclear facility is to be established and followed to demonstrate that the entire facility, especially items important for safety and radiation protection, have been constructed and function according to the design intent and to ensure that facility weaknesses are detected and addressed. While SRS reactors are old facilities and have been operated for a number of years, it is appropriate that the SRS reactors be evaluated against the intent of the Nuclear Safety Policy. A similar approach has been followed with regard to commercial nuclear standards as governed by "Initial Test Program for Water Cooled Nuclear Power Plants," NRC Regulatory Guide 1.68 (Reference 3).

8.2.2 Restart Criteria

The criteria for determining the adequacy of the SRS system testing/preoperational testing for restart are described below.

- (1) A master plan shall be issued to provide the guidelines for the development and conduct of the Startup Test Program (STP).
- (2) A Startup Test Program manual shall be issued to establish the processes and controls for the STP.

- (3) All systems determined to be evaluated by the STP shall be selected. The determination will be guided by the following criteria:
- (a) systems used for the shutdown and cooldown (cooling) of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period
 - (b) systems specified in NRC Regulatory Guide 1.68 that are applicable to the SRS reactors
 - (c) systems called upon by the newly-developed "symptom- based" Emergency Operating Procedures (EOPs) to maintain the vital safety functions of the plant during and after an emergency
 - (d) systems identified by the Probabilistic Risk Assessment (PRA) program as "critical features"
- (4) Restart Test Identification Reports (RTIR) shall be prepared for each of the systems identified in item 3 as being part of the Startup Test Program. Each RTIR shall include:
- (a) summary-level system functional description
 - (b) system functional requirements
 - (c) listing of known problem areas
 - (d) outline of the testing required to demonstrate compliance with the functional requirements
- (5) Test Procedures shall be prepared for all required tests when DPSOLs are not adequate and shall incorporate DPSOLs or parts of DPSOLs when necessary to provide an effective procedure for testing.
- (6) Tests shall be conducted as identified by the system RTIRs. For those tests identified as having been previously performed, the test results shall be documented and summarized, and the technical adequacy of the results shall be demonstrated.
- (7) A final test report shall be issued which summarizes the test results including the satisfactory disposition of test results that do not meet the restart criteria.
- (8) All completed test procedures, data, records, and reports shall be properly documented and stored.

8.2.3 Evaluation

In December 1990, the DOE staff reviewed WSRC's administrative control and operational procedures to determine the current effectiveness in fulfilling the System Testing/Preoperational Testing restart criteria. Each item below refers to the corresponding restart criteria.

- (1) A master plan was issued on August 15, 1989. This plan adequately provides guidelines for the development and conduct of the Startup Test Program (STP). It is titled "Startup Test Program Master Plan," OPS-STP-89001 (Reference 4). The master plan outlines the objectives, strategy, organization, administrative controls, scope, and schedule of the STP. In addition, the plan emphasizes that all program activities will be conducted in compliance with the "Savannah River Site Quality Assurance Manual," DPW 83-111-3 (Reference 5).

A special committee, the Joint Test Group (JTG), was specified in the master plan to provide independent review and approval of program activities. Membership of the JTG includes experienced personnel from Reactor Operations, Reactor Engineering, Maintenance, Reactor Quality Engineering, and the commercial nuclear industry. The DOE participates in JTG activities as a non-voting member. This board participation provides a key basis for a safe and effective startup program.

An ad hoc committee, the Startup Test Review Committee (STRC), was established in January 1990 to select the list of systems to be tested and to review the scope of testing developed by the STP. The DOE participates in the STRC activities as a non-voting member.

- (2) The Startup Test Program Manual (Reference 6) was completed and issued for implementation on April 11, 1990. The manual governs the conduct of the Startup Test Program. The Startup Test Program Manual (STPM) meets the stated criteria. The STPM is being revised to reflect WSRC improvements throughout the various areas covered by the STPM (i.e. organizational changes, change controls and approvals, etc.). Several of these changes are results from lessons learned as these programs progress.
- (3) WSRC has completed its system selection to determine the systems to be evaluated and tested. The criteria for selection are the same as that stated in the restart criteria.
- (4) RTIRs have been prepared for the selected systems. The scope of the RTIRs meets the restart criteria.
- (5) A total of 436 test procedures has been identified by the RTIR process for the restart effort. The STPM requires that test procedures be prepared for all required tests when DPSOLs are not adequate and that DPSOLs or parts of DPSOLs be incorporated, when necessary, to provide an effective procedure for testing.
- (6) The STPM requires that tests be conducted as identified by the RTIR. For those tests identified as having been previously performed, the test results will be documented and summarized and the technical adequacy of the results will be demonstrated. The STPM requires JTG approval for acceptance of prior test results.
- (7) During the test program, deviations from test restart criteria will be resolved before proceeding beyond the next appropriate specified hold

point per STPM-06, "Control of Testing." A final test report will be issued as required per the Startup Test Program Master Plan, which summarizes the test results including the satisfactory disposition of test results that do not meet the restart criteria. Evaluations for individual tests will be completed as the respective test evaluations are completed as required per the associated RTIR.

- (8) STPM-09 requires that all completed test procedures, data, records, and reports will be properly documented and stored. The scope of STPM-09 meets the restart criteria.

8.2.4 Conclusion

The staff has reviewed the System Testing/Preoperational Testing program and has determined that the procedures now in place meet the restart criteria.

8.2.5 References

1. WSRC, "Reactor Operations Management Plan" (ROMP) WSRC-RP-89-368, June 1989.
2. Office of Environment, Safety, and Health, Department of Energy, "Policy Statement on Nuclear Safety Objectives for DOE Facilities" N5480, May 1989.
3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.68, "Initial Test Program for Water Cooled Nuclear Power Plants," Revision 2, August 1978.
4. WSRC, "Startup Test Program Master Plan," OPS-STP-890001, August 15, 1989.
5. WSRC, "Savannah River Site Quality Assurance Manual," DPW 83-111-3.
6. WSRC, "Startup Test Program Manual," RD5.2, April 11, 1990.

8.3 Power Ascension Testing

8.3.1 Background

Two of the primary reasons for the extended shutdown of the Savannah River Site (SRS) production reactors were (1) a P-Reactor event in which an unfactored buildup of "poison" during a planned shutdown caused an unpredicted level of reactivity in the core upon reactor startup, and (2) the way in which the operating contractor handled this event.

Westinghouse Savannah River Company (WSRC) committed to developing a plan that will establish a controlled, formal, and deliberate program for K-Reactor restart and the subsequent reactor power ascension. This program will include very deliberate and controlled startup core testing relating to criticality, rod worth, temperature coefficients of reactivity, poison buildup, axial and radial power distributions.

8.3.2 Restart Criteria

The criteria for determining the adequacy of the SRS power ascension testing program for restart are described below.

(1) Power Ascension Program

- (a) Procedures shall be developed and approved for each operational change from initial criticality, through power ascension, to full power.
- (b) Neutron monitoring shall be in place that will adequately and reliably provide indications of the reactivity and neutronic behavior of the reactor core.
- (c) A deliberate plan for operational changes during power ascension shall be in place that ensures that no allowed operational configuration will result in any fuel design limits being exceeded.

(2) Criticality Testing

- (a) Critical configuration at zero power shall be established. WSRC shall identify the predicted critical configuration at zero power and the acceptable range of deviation. The test procedure shall state what actions shall be taken if the values are outside the allowable range (including root cause analysis, reporting requirements, and the cessation of further testing). The procedures shall also require these actions to be completed within a timely, identified period.
- (b) Critical configurations at various power levels shall be established. WSRC shall identify the predicted critical configurations at various power levels and the acceptable ranges of deviation. The test procedure shall state what actions shall

be taken if the values are outside the allowable ranges (including root cause analysis, reporting requirements, and the cessation of further testing, as appropriate). The procedures shall also require these actions to be completed within a timely, identified period.

(3) Reactivity Testing

- (a) The reactivity worths of each gang of control rods shall be determined at zero power. WSRC shall identify the predicted reactivity worths of each gang rod at zero power and the acceptable range of deviation. The test procedure shall state what actions shall be taken if the values are outside the allowable range (including root cause analysis, reporting requirements, and the cessation of further testing, as appropriate). The procedures shall also require these actions to be completed within a timely, identified period.
- (b) The overall temperature reactivity coefficient shall be determined. WSRC shall identify the predicted values and allowable range of deviation. The test procedure shall state actions to be taken if the values are outside the allowable range (including root cause analysis, reporting requirements, and cessation of further testing, as appropriate). The procedure shall also require these actions to be completed within a timely, identified period.
- (c) Equilibrium Xenon concentrations shall be determined at various power levels.

(4) Power Distributions

- (a) Radial power distributions shall be determined at various power levels and core configurations. WSRC shall identify the predicted radial power distributions at various power levels and core configurations and the allowable ranges of distributions. The test procedure shall state actions to be taken if the values are outside the allowable ranges (including root cause analysis, reporting requirements and cessation of further testing, as appropriate). The procedure shall also require these actions to be identified in a timely, identified period.
- (b) Axial power shapes shall be determined at various core configurations. WSRC shall identify the predicted axial power shapes at various core configurations and the allowable ranges. The test procedure shall state actions to be taken if the values are outside the allowable ranges (including root causes analysis, reporting requirements, and cessation of further testing, as appropriate). The procedure shall also require these actions to be completed in a timely, identified period.

8.3.3 Evaluation

The staff reviewed the Startup Test Program document (Reference 1), which contains a plan for the development and approval of procedures associated with startup and power ascension for K-Reactor. The staff determined that this document was acceptable. Subsequently, WSRC issued the Startup Test Program Manual (Reference 2), which establishes procedural requirements for test scope development, test writing, test conduct, and test closeout. The staff determined that the controls contained in the STPM were, in general, satisfactory.

Restart testing will be controlled by Reactor Special Procedures for Fuel Loading (Chargeback) and Power Ascension. The Fuel Load Procedure was written, submitted to the U.S. Department of Energy (DOE) Chief Test Engineer for approval, and obtained DOE K-Reactor Area Representative signature approval prior to commencing fuel loading.

The Power Ascension testing regime is described in Restart Test Identification Report (RTIR) #4 (Reference 3). Test procedures are under development and will be submitted to DOE for approval. This is an open item.

8.3.4 Conclusions

Based on the above evaluation, the DOE staff concludes that WSRC has prepared an acceptable plan for the development and approval of power ascension test procedures. The exact scope of power ascension testing and the techniques to be used for the K-Reactor Restart and Power Ascension Test Program are not completely developed at this time. The staff will review the procedures developed for power ascension testing when they have been completed and made available for review. The closure of the open item specified above will be adequate to satisfy the restart criteria.

8.3.5 References

1. WSRC, RTR-2710, "K-Reactor Restart Strategy and Power Ascension Program", February 16, 1990.
2. WSRC Reactor Restart Division, "Startup Test Program Manual (RD5.2)," May 24, 1990.
3. WSRC, WSRC-TR-90-29-004, "RTIR #4, Power Ascension Restart Test Identification Report," April 1990.

9 SAFETY REVIEWS

9.1 Operational Safety Review Process

9.1.1 Background

In order to ensure that a high degree of station nuclear safety is maintained, operational safety review groups have been established at commercial nuclear facilities to advise station managers on safety matters. The individuals who make up these groups should have diversified backgrounds that enable them to apply a broad perspective of expertise to safety review matters brought before them.

An operational safety review group remains independent of any single line organization group. At the time of shutdown, such a committee independent of any single line organization did not exist to oversee nuclear safety at the Savannah River Site (SRS) reactors.

9.1.2 Restart Criteria

(1) General

- (a) A reactor operational safety review committee shall be established and initiated to conduct onsite safety review of plant operational activities, review reactor operation policies and procedures, perform safety evaluations of reactor operational events and incidents, provide guidance and assistance in the understanding and interpretation of plant Technical Specifications and other operations related requirements, and perform analysis of reactor performance trends. The committee shall consist of individuals having, as a minimum, diversified backgrounds in the areas of reactor operations, maintenance, engineering, and radiological protection. Committee permanent members and alternates must possess sufficient education, experience, expertise, and training to undertake the reviews and evaluations which the committee is intended to perform. Permanent and alternate committee members shall also be trained in the functions, duties, and responsibilities of the overall committee and of their respective roles on the committee. In the performance of safety evaluations, committee members shall be knowledgeable of:
 - o the definition of an Unreviewed Safety Question, as delineated in DOE Order 5480.6 (Reference 1)
 - o the technical aspects of the reactor systems being reviewed
 - o the contents of the reactor Safety Analysis Report and the plant Technical Specifications
 - o the significance of internal and external events that could have the potential for affecting plant safety

- (b) Committee members shall be retrained periodically to strengthen their review abilities. Such training shall focus on internal and external events that have occurred since the last training received, on lessons learned from those events, and with respect to new technical developments and their relationships to reactor design and operation.

(2) Committee Charter

A charter which describes how the operational safety review committee will function shall be formalized and address the following:

- o membership, appointment of alternates, and quorum requirements
- o qualification and training requirements for permanent and alternate members
- o methods for executing responsibilities
- o use of subcommittees or "qualified reviewers"
- o documentation of reviews (meeting minutes), discussions, and dissenting opinions
- o interfaces with an oversight safety review group

(3) Operational Reviews

The operational safety review committee shall be responsible for reviewing the following:

- o reactor trips and associated evaluations
- o selected performance monitoring trends
- o trend reports resulting from quality assurance and surveillance program activities
- o assessments or reports concerning the conduct of selected station activities
- o evaluations of significant internal and external operating experience information related to nuclear safety and recommended actions to prevent occurrence or recurrence
- o investigations of all Technical Specification violations, including the preparation and forwarding of report evaluations and recommendations to prevent recurrence
- o Unusual Occurrence Reports
- o plant operations to detect potential hazards to safety

- o any accidental, unplanned, or uncontrolled radioactive releases
- o all proposed tests and experiments that affect nuclear safety; not applicable to Startup Test Program where a Joint Test Group (JTG) is in place
- o proposed new procedures, procedure changes or temporary procedure changes which may involve an unreviewed safety question
- o all proposed Technical Specifications changes
- o any proposed changes or modifications to plant systems or equipment that may affect nuclear safety
- o special reviews, investigations, or analyses as requested by the Deputy General Manager for Reactor Operations (DGMRO), or as requested by the Reactor Independent Safety Review Committee (RISRC) (see SER Section 9.2), and prepare reports on the results of those special reviews, investigations or analyses

(4) Committee Interfaces

Interfaces between the operational safety review committee and the RISRC (see SER Section 9.2) shall be clearly defined as to:

- o review by the RISRC of the documentation generated by the operational safety review committee
- o operational safety review committee representation at meetings conducted by the RISRC
- o periodic assessment of operational safety review committee meeting activities by the RISRC
- o feedback of the RISRC assessments of operational safety review group activities

9.1.3 Evaluation

RDP 2.04, "Reactor Operations Safety Committee" (ROSC) - (U), Revision 0 (Reference 2), specifies the requirements for staffing the ROSC. The procedure lists specific diversified backgrounds for the committee members and the specific functional areas that are to be represented. Section 4.1.1 of the procedure specifies that the ROSC members "...must possess sufficient education, experience, expertise, SAR and technical training to undertake the reviews and evaluations which the committee is intended to perform. Six years of power plant and/or SRS experience with four years being in a nuclear power plant is the minimum experience requirement." ROSC members are specified by position. The positions specified are senior management positions that require the type of education, experience, etc. required by the SER restart criteria.

RDP 2.04, Section 3.0, delineates the responsibilities of the ROSC in general and of the Chairman and Secretary of the ROSC, in particular. Specifically, the ROSC is tasked with thoroughly evaluating each activity brought before the committee to determine that nuclear-safety-related aspects and operability/maintainability aspects of the facility are being adequately considered. The chairman and secretary of the ROSC are responsible for conducting and documenting the meetings in accordance with RDP 2.04.

RDP 2.04, Section 4.1.2, requires that, upon initial appointment and on a biannual basis, the committee members read and review the ROSC procedure, review the Technical Specifications, and read and review other material designated by the ROSC chairman. Upon initial appointment, each member must successfully complete a course on the evaluation of Unreviewed Safety Questions. This section also requires that periodic presentations be made to the ROSC on internal and external events that have occurred since the last presentation, on lessons learned from those events, and on new technical developments and their relationships to reactor design and operation.

RDP 2.04 serves as the charter governing the activities of the ROSC. RDP 2.04 Section 1.0, "Purpose," states that "The Reactor Operations Safety Committee (ROSC) conducts safety reviews of plant operational activities...." Section 4.1 discusses the qualification and training for ROSC members and the appointment of members including alternates. Section 4.3 discusses the quorum requirements and delegation responsibilities of the ROSC and its members. The training requirements are outlined in paragraphs 4.1.1 and 4.1.2. Sections 4.4, 4.5, and 5.0, discuss the conduct of ROSC meetings, the documentation requirements, and the retention requirements for ROSC-generated documentation.

The items discussed concerning conduct of meetings include frequency of meetings, timetables for distribution and review of documents for the meetings, and mechanisms for placing items on the agenda. The documentation required from the ROSC meetings includes meeting minutes, evaluations of the items discussed, any action items assigned, and any outstanding items requiring further discussion. The ROSC secretary submits meeting agendas, reviewed documents, and meeting minutes to Records Management. In addition, ROSC meeting attendance sheets, reading lists and their signoffs, and qualification/training records are generated by the procedure and are maintained and disposed of in accordance with the Reactor Operations Records Retention Schedule.

RDP 2.04, Section 4.2 describes the various items requiring review by the ROSC. Among these items are reactor trips and associated evaluations; selected performance monitoring trends; trend reports resulting from quality assurance and surveillance program activities; and evaluations of significant internal and external operating experience information related to nuclear safety and recommended actions to prevent occurrence or recurrence. The ROSC must perform special reviews, investigations, analyses, and reports as requested by the DGMRO or by the RISRC as required by Section 4.2.3 and must provide guidance and assistance in understanding the Technical Specifications. Section 4.2.1(1) requires the ROSC to review proposed revisions to (1) procedures that establish administrative controls to implement the QA program within Reactor Operations Department and the Technical Specifications

surveillance program; (2) normal, abnormal, and emergency operating procedures; (3) procedures for controlling and responding to effluent releases potentially having radiological consequences; and (4) fuel and target handling procedures. Review of items (1) through (4) includes temporary procedure changes. Section 4.2.1(2) requires the committee to review reportable events, such as Unusual Occurrence Reports (UORs) and violations of the Technical Specifications, including the review of investigations and recommendations for preventing or reducing recurrences of the event in question. The Operations and Maintenance Procedure Review and Approval Committee (OMPRAC) is a subcommittee of the ROSC and performs the procedure review and approval duties of the ROSC in accordance with RD 7.1 DMP 7.02, "Operations and Maintenance Procedure Review and Approval Committee" (OMPRAC) - (U), Revision 1 (Reference 3). Details of the procedure review process can be found in SER Section 10.3.

In addition to the items mentioned above, the ROSC is also responsible for reviewing the following items in accordance with RDP 2.04 Section 4.2.1:

- o any proposed changes to plant systems or equipment that may affect nuclear safety
- o proposed changes to the Technical Specifications
- o Reactor Facility Emergency Plans and implementing procedures
- o Fire Protection Program and implementing procedures
- o plant operations to detect potential hazards to safety
- o all proposed tests and experiments that affect nuclear safety except those applicable to the startup test program under the cognizance of a JTG
- o accidental, unplanned, or uncontrolled radioactivity releases
- o reactor operation policies and procedures

RDP 2.04 Section 4.5.1 describes the interface requirements between the ROSC and the RISRC. Section 4.5.1 states that meeting minutes from the ROSC meetings will be distributed to the DGMRO and the RISRC. Section 7 of RDP 2.04 describes the RISRC as a higher-tier safety review committee than the ROSC, which provides overview of the ROSC and other matters as outlined in the Technical Specifications. The ROSC is also responsible for forwarding recommendations or requests for assistance to the RISRC on specific items identified by the ROSC chairman.

Section 2.0 of RDP 2.04 states that the ROSC chairman will present all ROSC findings to the DGMRO and the RISRC. Section 4.5.1 requires the ROSC chairman to distribute copies of the meeting minutes to the DGMRO and the RISRC within 14 calendar days following the meeting. In addition, RDP 1.09, "Reactor Independent Safety Review Committee" (RISRC) - (U) Draft Revision O-A, (Reference 4) requires that the RISRC review reports, meeting minutes, and other records of the activities of the ROSC and provide recommendations as

deemed appropriate. The restart criteria concerning RISRC reviews of specified ROSC generated documentation are met by RDP 1.09, but since the procedure is not yet effective, this issue remains an open item.

RDP 1.09, Section 4.4.5, the RISRC procedure quorum requirements, do not specify that an ROSC member be present at the RISRC meetings as specified by SER Restart Criteria 9. Pending revision to RDP 1.09 to include this requirement, this issue remains an open item.

9.1.4 Conclusion

With the exception of the two open items identified above, the operational safety review process as discussed in this evaluation meets the requirements of the restart criteria. The resolution of these items will be addressed in a supplement to the SER.

9.1.5 References

1. U. S. Department of Energy, DOE Order 5480.6, "Safety of DOE-Owned Nuclear Reactors."
2. WSRC, RDP 2.04, "Reactor Operations Safety Committee" (ROSC) - (U) Revision 0.
3. WSRC, RD 7.1, DMP 7.02, "Operations and Maintenance Procedure Review and Approval Committee" (OMPRAC) -(U), Revision 1.
4. WSRC, RDP 1.09, "Reactor Independent Safety Review Committee" (RISRC) - (U) (Draft 0-A).

9.2 Independent Safety Review Process

9.2.1 Background

The independent safety review process provides oversight of activities, and of the programs for managing these activities, to ensure compliance with DOE Orders and ensure enhanced reliability and safety.

A safety review committee at the senior management level is an important element of an overall independent safety review program for nuclear facilities. The Reactor Safety Advisory Committee (RSAC) was chartered in 1982 as a result of concerns raised by the 1981 report, "Safety Assessment of DOE Nuclear Reactors" (Reference 1). The purpose of the RSAC is to provide independent overview of the SRS reactors and nuclear operations. The Committee is comprised of seven senior-level nuclear industry consultants with diverse technical backgrounds.

In 1985, based on a RSAC recommendation to strengthen safety assessments by providing a clearly independent and dedicated, multidiscipline group, the Reactor Safety Evaluation Section (RSES) was formed. The RSES reviews and concurs with reactor safety documentation, such as Safety Analysis Reports (SARs), Technical Specifications, Test Authorizations, selected design changes, and WSRC corporate-level safety concerns or incidents; monitoring of nuclear industry events and programmatic information to determine applicability to SRS activities and disseminating applicable information to responsible management; and performing independent oversight to ensure compliance with DOE Orders.

WSRC also has established a Reactor Independent Safety Review Committee (RISRC) to provide for senior management oversight of the reactor safety review system. The RISRC is modeled after the Corporate Safety Review Committee addressed in American National Standards Institute/American Nuclear Society (ANSI/ANS) standard 3.2-1988 (Reference 2). The goal of this programmatic change is to more closely align the reactor safety review program with similar programs found in the commercial nuclear industry. The RISRC will be established prior to restart. The function of the RISRC is to provide an independent review of unreviewed safety question determinations (USQDs), as discussed in section 5.2, and an assessment of the results of audits of designated activities in the areas listed in the following restart criteria.

9.2.2 Restart Criteria

The criteria for determining the adequacy of the SRS independent review process for restart are described below.

(1) General

- (a) An independent safety review committee shall be established to provide senior management the capability to oversee and assess the nuclear safety aspects of reactor activities.

- (b) The committee's function and responsibilities shall be delineated in the Administrative Controls section of the Technical Specifications.
- (c) The committee purpose, objectives, functions, authority, responsibility, composition, quorum, meeting frequency and reporting requirements shall be clearly defined and delineated as specified in Section 4.4 of ANSI/ANS 3.2-1988, "Administrative Controls and Quality Assurance for the Operation Phase of Nuclear Power Plants," and in Section 8.g. of U.S. Department of Energy Order 5480.6 (Reference 3). The committee's charter shall provide guidance for all requirements referenced in the Technical Specifications.

(2) Function

- (a) The committee shall provide independent review of activities in the areas of:
 - o nuclear reactor operations
 - o nuclear engineering
 - o chemistry and radiochemistry
 - o metallurgy
 - o instrumentation and control
 - o radiological safety
 - o mechanical and electrical engineering
 - o quality assurance practices
- (b) The committee members and alternates shall be independent from the pressures of reactor operations and shall report to a level of senior management sufficiently high enough to ensure that recommended corrective actions are taken.

(3) Composition

The committee shall be composed of at least five individuals, including the chairman. The chairman shall be appointed in writing by the President of WSRC. Members, including designated alternates, shall be formally recommended by the chairman and appointed in writing by the President WSRC. Each member shall meet the qualifications specified in Section 4.7 of ANSI/ANS-3.1-1987 (Reference 4).

(4) Alternates

Alternate members shall be appointed in writing by the President of WSRC to serve on a temporary basis; however, no more than a minority shall participate as voting members in committee activities at any one time.

(5) Consultants

Consultants shall be used as determined by the committee chairman to provide expert advice to the committee.

(6) Meeting Frequency

The committee shall meet at least once per calendar quarter during the initial year of plant operation following restart, and at least once per 6 months thereafter.

(7) Quorum

The quorum necessary for performance of the committee review and audit functions shall consist of the chairman or his/her designated alternate, and at least of majority of regular committee members. No more than a minority of the quorum shall have line responsibility for operation of the reactors.

(8) Unreviewed Safety Questions and Technical Specifications

The committee shall review applicable documents and programs to ensure that plant activities are conducted safely and do not constitute unreviewed safety questions or Technical Specification violations as defined in restart criteria Sections 5.2 and 11.2.

(9) Reviews

The committee shall be responsible for the review of:

- o Safety evaluations for (1) changes to procedures, equipment, or systems; and (2) tests or experiments to verify that such actions did not constitute an unreviewed safety question.
- o Proposed changes to procedures, equipment, or systems that involve an unreviewed safety question.
- o Proposed tests or experiments that involve an unreviewed safety question.
- o Proposed changes to Technical Specifications.
- o Violations of Codes, regulations, orders, Technical Specifications, or of internal procedures or instructions having nuclear safety significance.
- o Significant operating abnormalities or deviations from normal and expected performance of station equipment that affect nuclear safety.
- o All Unusual Occurrence Reports (UORs).
- o All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.
- o Reports, meeting minutes, and activities of the Plant Nuclear Safety Committee (PNSC) ensuring, as a minimum, that:

- The quality and effectiveness of safety evaluations is maintained.
- Appropriate activities and events are subject to review.
- Reviews are sufficient in scope and depth to identify and address issues involved.
- All positions and concerns are considered reviewed and dispositioned.
- Appropriate conclusions are reached based on an adequate technical analysis.
- Action items are tracked through to completion, and that actions are completed in a timely manner.

(10) Records

Records of committee activities shall be prepared and distributed as follows:

- o Minutes of each meeting shall be prepared and forwarded to the WSRC President within 14 days following each meeting.
- o Reports of required reviews shall be included in the minutes where applicable or forwarded under separate cover to the WSRC President within 14 days following completion of the review.
- o Audit reports shall be forwarded to the WSRC President and to the management positions responsible for the areas audited within 30 days after completion of the audit.

(11) Audits

Independent audits of reactor operation activities shall be performed under the cognizance of the committee. These audits shall be performed at least once within a 12 month period. The audits shall encompass as a minimum:

- o The conformance of reactor operations to the Technical specifications.
- o The performance, training, and qualifications of reactor operations personnel.
- o The results of actions taken to correct deficiencies occurring in reactor equipment, structures, systems, or methods of operation that affect nuclear safety.
- o The performance of activities required by the Quality Assurance Program to meet the criteria of ANSI/ASME NQA-1-1986.

- o The fire protection programmatic controls including the implementing procedures, the fire protection equipment and program implementation. An outside independent fire protection consultant shall be used at least every third year.
- o The Radiological Environmental Monitoring Program and resulting issues.
- o The Emergency Plan and implementing procedures.
- o The Offsite Dose Calculation Manual, or equivalent document, and implementing procedures.
- o The Process Control Program and implementing procedures for processing and packaging of radioactive wastes.
- o The performance of activities required by the QA Program for effluent and environmental monitoring.
- o The Security Plan and implementing procedures.
- o Any other area or site activity considered appropriate by the President, WSRC.

9.2.3 Evaluation

Revision 0 of the Technical Specifications (Reference 5), approved by DOE on January 25, 1991, includes the Administrative Control Technical Specifications. The Administrative Control Technical Specifications (Section 5.7.2) explicitly delineate the RISRC function and responsibilities, designating the areas of review and audit activities as well as listing specific activities, events, or documents requiring review as described in the restart criteria. These areas concur with the areas listed in the restart criteria; however, DOE Order 5480.6, Section 8.g (8), which is applicable to the restart criteria, lists other responsibilities for review.

The Administrative Control Technical Specifications were revised to fully meet the requirements of DOE Order 5480.6, Section 8.g (8) in the areas of (1) administrative procedures, (2) organization and staffing, (3) appropriate training programs, and (4) initial and subsequent qualification and certification requirements and procedures. The Administrative Control Technical Specifications do not require that members, including designated alternates, be recommended by the Chairman and appointed in writing by the President of WSRC. In discussions between senior DOE and WSRC representatives, it was agreed that the Vice President and General Manager, Reactor Restart Division is an appropriate level for committee appointments and reporting. This exception to the above restart criterion is consistent with the practices at commercial nuclear reactors and is acceptable.

The Administrative Control Technical Specifications (Section 5.7.2.6) provide for committee independence from the pressure of operations while including access to senior management, as required by the restart criteria.

An implementation plan for the Administrative Control Technical Specifications was approved by DOE on November 16, 1990. The plan calls for full implementation by restart. Many activities including the independent review of USQDs have already been implemented.

9.2.4 Conclusions

Implementation of the Administrative Control Technical Specifications will be monitored by the DOE staff and documented in the monthly reports. There are no open items.

9.2.5 References

1. U. S. Department of Energy, "Safety Assessment of DOE Nuclear Reactors" (1981).
2. ANSI/ANS Standard 3.2-1988, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," American National Standards Institute (1988).
3. DOE Order 5480.6, "Safety of U.S. Department of Energy-Owned Reactors," September 23, 1986.
4. ANSI/ANS Standard 3.1-1987, "Selection, Qualification and Training of Personnel for Nuclear Power Plants," American National Standards Institute (1987).
5. "Technical Specifications WSRC-TS-10003," Revision 0 (U), December 18, 1990.

9.3 Post-Trip Assessment

9.3.1 Background

WSRC recognizes the safety significance of reactor trips and is committed to the development and implementation of a systematic method for diagnosing the root cause(s) of a reactor trip, ascertaining the proper functioning of safety-related and other important equipment during the trip, and making a determination about whether the plant can be restarted safely. The post-trip assessment process is based on Institute of Nuclear Power Operations (INPO) report INPO 88-024, "Post-Trip Reviews," (Reference 1) and on American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 3.2, 1988, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants" (Reference 2).

The post-trip assessment process consists of two major areas: (1) organization and administration; and (2) oversight and review. WSRC has programs in place that address organization and administration as well as post-trip assessment reviews. The applicable programs include Reactor Restart Division Administrative Procedure (RDP) 1.09, "Reactor Independent Safety Review Committee (RISRC)" (Reference 3); Reactor Department Standing Instruction (ROSI) II.16, "Reactor Scram/Unscheduled Shutdown Recovery" (Reference 4); RDP 12.02, "Notification of Reactor Plant Conditions, Events, and Concerns" (Reference 5); RDP 2.04, "Reactor Operations Safety Committee (ROSC)" (Reference 6); and DPSOL 105-1205A, "Post-Shutdown Data Collection and Review" (Reference 7). These programs and procedures have been upgraded or developed since April 1, 1989.

9.3.2 Restart Criteria

The criteria for determining the adequacy of the SRS post-trip assessment program for restart are described below.

(1) Post-Trip Assessment Organization and Administration

- (a) The post-trip review process shall be clearly defined.
- (b) Responsibilities of interfacing organizations, both internal and external, shall be proceduralized and understood.
- (c) Staffing and resources shall be sufficient to accomplish assigned tasks.
- (d) Post-trip review procedures shall specify the responsible personnel and the actions to be completed with respect to:
 - o data collection
 - o trip investigation
 - o classification of events
 - o restart decisions
 - o determination of maintenance and/or testing required for restart

- o trip investigation review
 - o identification of lessons learned and follow-up actions
 - o interface with Operating Experience Review program
 - o interface with oversight and review organizations
- (e) Post-trip reviews shall be comprehensive and shall include the following:
- o identification and resolution of the root cause(s) of the trip through a formal documented process
 - o identification and resolution of discrepancies between actual and expected plant responses to the trip
 - o documentation to support results and recommendations
- (2) Post-Trip Assessment Oversight and Review
- (a) Post-trip reviews shall be assessed for accuracy, root cause(s) determination, and corrective action measures by an organization not responsible to line management.
 - (b) An operations review committee shall review plant-trip reports.
 - (c) The operational safety review group (see section 9.1 of this SER) shall review the plant-trip report prior to restart if (1) the cause of the reactor trip is not positively known, or (2) safety-related or other important equipment functioned in an abnormal or degraded manner during the trip, and the malfunction has not been corrected or redundant equipment is not available to allow start-up.
 - (d) An oversight nuclear safety group shall review plant trip reports.

9.3.3 Evaluation

In January 1991 the DOE staff reviewed WSRC's administrative control and operational procedures to determine the current effectiveness in fulfilling the post-trip assessment restart criteria. The results of the staff's evaluation are discussed below.

Post-Trip Assessment Organization and Administration

The post-trip review process by line management is clearly defined in DPSOL 105-1205A, Rev. 2 and DPSOP 336, II.16, Rev. 5. The responsibilities of operational and independent interfacing organizations are proceduralized in DPSOL 105-1205A, Rev. 2 and in DPSOP 336, II.16, Rev. 5. The independent organizations include WSRC Reactor Safety Evaluation Section, WSRC Reactor Engineering, and DOE-SR.

A review of organizational charts indicates that staffing and resources are sufficient to accomplish assigned tasks. Post-trip review procedures specify most responsible personnel and most actions to be completed. The Plant Manager

is responsible for data collection and for initiation of the trip investigation. The Reactor Technology (RT) engineer-on-call is responsible for specifying the additional information necessary to determine the root cause(s) of the trip. Classification of the trip as Condition I or Condition II is determined by the Shift Manager, with concurrence from the RT engineer-on-call, and is based on the results of the investigation, performance of DPSOL 105-1205A, and discussions with the Plant Manager. The Deputy General Manager, Reactor Operations will approve reactor restart after obtaining concurrence from the Managers of DOE Reactor Operations, WSRC Reactor Operations, WSRC Reactor Engineering, and the WSRC Reactor Safety Evaluation Section. Nonconcurrence by any of these managers will prohibit reactor restart.

Post-trip review procedures do not specify the responsible personnel and actions to be completed for determination of maintenance and/or testing required for restart. This issue is an open item.

Post-trip investigation review is accomplished by the Managers of DOE Reactor Operations, WSRC Reactor Operations, WSRC Reactor Engineering, and WSRC Reactor Safety Evaluation Section (in accordance with DPSOP 336, II.16, Rev. 5) prior to concurring in reactor restart. Follow-up actions are identified in the Reactor Event Report (RER), which is required for all unscheduled shutdowns, in accordance with Reactor Engineering Department Training and Operating Guide, Section 1, "Reactor Event Reporting System" (Reference 9). Lessons learned that could be of importance or that should be addressed in personnel training or facilities procedures are identified in the final RER.

Post-trip procedures do not specify the responsible personnel and actions to be completed with respect to interface with the Operating Experience Review program. This issue is an open item.

Interface with oversight and review organizations is specified in post-trip procedures. DPSOP 336, II.16 requires the concurrence of three oversight and review organizations prior to restart. In addition, the Reactor Engineering Manager has the option of convening the Reactor Event Review Committee to concur in the post-trip investigation.

Identification and resolution of the root cause(s) of the trip through a formal documented process are not required during the post-trip investigation. This issue is an open item.

Identification and resolution of discrepancies between actual and expected plant responses to the trip are not required during the post-trip investigation. This issue is an open item.

Because identification and resolution of the root cause(s) of the trip is not accomplished through a formal documented process, and discrepancies between actual and expected plant responses to the trip are not required, sufficient documentation to support results and recommendations is not provided for. This issue is an open item.

Post-Trip Assessment Oversight and Review

Post-trip review by an organization not responsible to line management is provided for in RDP 1.09 "Reactor Independent Safety Review Committee (RISRC)." The RISRC provides advice, expertise and independent review of designated activities with formal reporting requirements to the Vice President and General Manager, Reactor Restart Division. The RISRC ensures that selected reports, meeting minutes, and other activities of the Reactor Operations Safety Committee are reviewed. The RISRC also ensures that the quality and effectiveness of evaluations are maintained, appropriate topics are reviewed, and reviews are sufficient in scope and depth to identify and address issues involved.

The RISRC ensures that all positions and concerns are considered, reviewed, and dispositioned; appropriate conclusions are reached, based on adequate technical analysis; action items are tracked through to completion; and actions are completed in a timely manner.

An operations review committee reviews plant-trip reports in accordance with RDP 2.04 "Reactor Operations Safety Committee (ROSC)." Minutes of all ROSC meetings are sent to the RISRC for review. In addition the ROSC chairman may request assistance from the RISRC.

The ROSC is not required to review the plant-trip report prior to restart if (1) the cause of the reactor trip is not positively known, or (2) safety-related or other important equipment functioned in an abnormal or degraded manner during the trip, and the malfunction has not been corrected or redundant equipment is not available to allow start-up. This issue is an open item.

The RISRC provides senior management oversight of the reactor safety review system. However, it is not clear that the RISRC is required to review all plant-trip reports. This issue is an open item.

9.3.4 Conclusion

Several open items remain to be resolved before the staff can conclude that post-trip assessment has been adequately addressed. Resolution of these open items will be documented in a supplemental to this SER.

9.3.5 References

1. INPO 88-024, "Post-Trip Reviews," December 1988.
2. ANSI/ANS 3.2, 1988, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," April 6, 1989.
3. Reactor Restart Division Administrative Manual, RDP 1.09, Rev. 0, "Reactor Independent Safety Review Committee (RISRC)," February 11, 1991.

4. DPSOP 336, ROSI II.16, Rev. 5, "Reactor Scram/Unscheduled Shutdown Recovery," September 24, 1989.
5. Reactor Restart Division Administrative Manual, RDP 12.02, Rev. 1, "Notification of Reactor Plant Conditions, Events, and Concerns," December 12, 1990.
6. Reactor Restart Division Administrative Manual, RDP 2.04, Rev. 0, "Reactor Operations Safety Committee," July 28, 1990.
7. DPSOL 105-1205A, Rev. 2, "Post-Shutdown Data Collection and Review," February 7, 1990.
8. Reactor Engineering Department Training and Operational Guide, Section I, "Reactor Event Reporting System," March 1, 1989.
9. Reactor Restart Division Administrative Manual, RDP 9.02, Rev. 2, "Occurrence Reporting System," January 1, 1991.

10 OPERATIONS

10.1 Staffing and Qualifications

10.1.1 Background

Proper selection, qualification, and training of personnel are significant factors in ensuring the safe, reliable and efficient operation of reactor facilities. Because the cost to train personnel is a significant resource investment, it should be predicated on a reasonable probability of success. Properly defined education and experience standards are designed to provide that level of assurance that individuals hired are capable of successfully completing the training program. The training and qualification programs are designed to assure that only individuals who are capable of safely performing their assigned duties are authorized to operate the reactor facilities.

This section and Section 10.2, "Training," outline the minimum requirements for both the selection and training of selected reactor facility personnel. These criteria are based on commercial industry standards such as American National Standards Institute (ANSI)/American Nuclear Standards (ANS) 3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants" (Reference 1) and DOE Order 5480.6, "Safety of DOE-Owned Reactors" (Reference 2). This section also discusses the minimum staffing levels designed to assure that the reactor is operated with adequate numbers of qualified personnel to assure safe operations.

One of the major issues contributing to the prolonged shutdown of the K-Reactor following its routine refueling outage in April 1988 was the performance of personnel in their conduct of operations and support duties. After the Three Mile Island (TMI) accident in 1979, significant attention and resources were devoted to upgrading the technical qualifications of commercial operators. Although increased attention was focused on Savannah River Site (SRS) programs as well, the rate and effectiveness of these changes did not keep pace with the commercial sector. The fact that SRS training does not meet industry standards is a driving force behind the new push for excellence. The need for and desire to obtain increased operator knowledge levels is recognized and strongly endorsed by WSRC management. This fact alone represents a significant cultural change in the manner in which the reactors are operated and training conducted.

Now that Westinghouse Savannah River Company (WSRC) operations and plant management acknowledges and supports the need for training improvements and insists the upgrades begin in the near-term, the process of training can improve. Staffing and management changes in the training department have resulted in significantly increased awareness of commercial and naval training standards and practices. Numerous short-term and long-term programs have been implemented to address the weaknesses in operator knowledge; however, training of this magnitude is not a short-term solution, and compensating measures are necessary until such time as long-term programs become effective. This section and Section 10.2 discuss specific weaknesses and the compensating measures being taken.

10.1.2 Restart Criteria

The criteria for determining the adequacy of staffing and qualifications at SRS for restart are described below.

(1) General

As specified in both the draft DOE Nuclear Safety Policy and Policy and Expectations for Formal Conduct of Operations in DOE Owned Nuclear Material Production Program Facilities, facilities shall be operated and supported by personnel designated by management as qualified to perform their assigned function. Minimum qualification criteria shall be established and documented consistent with the requirements in DOE Order 5480.6; ANSI/ANS 3.1; draft DOE Order 5480.XX, "Personnel Section, Qualification, and Training Requirements" (Reference 3); and the NRC Policy Statement on Education for Senior Operators and Shift Supervisors at Nuclear Power Plants. Deviations from these requirements will require the approval of the Department of Energy Savannah River Site. Selected positions and their associated requirements are highlighted due to the restart issues that surround them.

(2) Education

Educational criteria are established to ensure personnel possess the prerequisite academic skills required to successfully complete training classes. Training of operations and support personnel is a significant resource investment which should be predicated on a reasonable probability of success. Personnel assigned to their positions prior to June 1, 1989, who do not meet the new acceptance criteria for the position are not disqualified from continuing to function in those positions provided they continue to meet the training and examination requirements provided for their positions. Acceptable alternatives to educational requirements include the following:

- o A GED may be used in lieu of a High School Diploma.
- o Successful completion of technically oriented post- secondary courses may be credited even if a degree was not obtained.

(a) Reactor Building Operator (Non-certified)

Minimum education requirements for selection into the position of Reactor Building Operator shall be a high school diploma or equivalent unless approved otherwise by DOE on an individual basis. Goal recruiting criteria should be an Associate Degree from an accredited university or college in a technical area.

(b) Central Control Room Operator (Certified)

Minimum education requirements for selection into the position of Central Control Room Operator shall be a high

school diploma or equivalent unless approved otherwise by DOE on an individual basis. Goal recruiting criteria should be an Associate Degree from an accredited university or college in a technical area.

(c) Supervisor/Central Control Room Supervisor (Certified)

Minimum education requirements for selection into the position of Supervisor shall be a high school diploma or equivalent unless approved otherwise by DOE on an individual basis. At least one supervisor per shift shall have a Bachelor of Science (BS) Degree from an accredited university or college in an engineering or related science field. A degreed certified supervisor is to function as the Shift Technical Engineer (STE).

(d) Support Staff Supervisor (Non-Certified)

Minimum education requirements for selection into the position of Supervisor shall be a high school diploma or equivalent.

(e) Maintenance Personnel (Mechanical and Electrical)

Minimum education requirements for selection into the position of mechanic or technician shall be a high school diploma or equivalent.

(f) Plant Technical Support Personnel (Engineering)

Minimum education requirements for selection into the plant technical support engineering staff shall be a Bachelor of Science degree from an accredited university or college in an engineering or related science field.

(g) Components Handling Reactor Operator (CH-RO) (Non-certified)

Minimum education requirements for selection into the position of CH-RO shall be a high school diploma or equivalent.

(h) Components Handling Senior Reactor Operator (CH-SRO) (Certified)

Minimum education requirements for selection into the position of CH-SRO shall be a high school diploma or equivalent.

(i) Components Handling Supervisor (Certified)

Minimum education requirements for selection into the position of Supervisor shall be a high school diploma or

equivalent.

(j) Components Handling Supervisor (Non-Certified)

Minimum education requirements for selection into the position of Supervisor shall be a high school diploma or equivalent.

(3) Experience

Experience requirements are established to ensure personnel possess the knowledge required to safely conduct the tasks assigned, and that this knowledge can be practically applied. Reactor experience is defined as direct participation in reactor design, construction, startup testing, operations, maintenance, or technical services. Observation of these activities is not considered experience. Personnel assigned to their positions prior to June 1, 1989, or entered training for their positions prior to June 1, 1989, and do not meet the selection criteria for the positions are not disqualified from continuing to function in those positions provided they continue to meet the training and examinations requirements for their positions. Acceptable alternatives to experience requirements include:

- o Post-secondary education in related technical fields from accredited universities or colleges may be substituted for experience requirements on a one for two basis (e.g., 1 year of experience credit for a 2 year Associate degree, 2 years credit for a BS) up to 50 percent of the total experience requirement.
- o Onsite on-the-job training (OJT) is credited as experience on a one for one basis up to 2 years maximum.
- o Initial classroom training regarding position duties and knowledge requirements may be credited for experience requirements on a one for two basis up to 9 months experience.
- o Periods of prolonged shutdown (greater than 1 year) are credited on a 75 percent basis within 1.5 years of restart and 50 percent basis for time more than 1.5 years prior to restart. This credit is limited to 1 year of each or 2 years combined.
- o Structured simulator exercises may be credited on a five to one basis (e.g., 5 weeks of experience credit for a 1 week simulator exercise) up to 1 year of credit if conducted on a reference plant simulator and three to one basis for a similar plant simulator. The SRS simulator may be considered a reference plant simulator for P, L, and K reactors.
- o Experience at P-, L-, or K-Reactor may be credited on a one for one basis at any of the other two reactors.

- o Commercial nuclear power experience may be credited on a 75 percent basis with no limit and military power plant experience on a 50 percent basis up to 3 years.

- (a) Reactor Building Operator (RBO)

An RBO shall have 1 year of reactor experience, including 6 months onsite, prior to assuming independent duties.
(Exception: Those personnel selected into the RBO program prior to January 1, 1990, were selected based upon the minimum educational criteria and placed in training programs which were based on 6 months of experience prior to qualification.)

- (b) Central Control Room Operator (CCRO)

A CCRO shall have 3 years of reactor experience, including 6 months onsite, prior to assuming independent duties.

- (c) Supervisor/Central Control Room Supervisor (CCRS) - Certified

A certified supervisor shall have 3 years of reactor experience, including 6 months onsite, prior to assuming independent shift duties. In addition, the control room supervisor shall have a minimum of 6 weeks experience at an SRS reactor facility operating above 20 percent rated power. A supervisor serving as a Shift Technical Engineer may have waived the requirement for 3 years of reactor experience and 6 weeks of power experience.

- (d) Supervisor - Non-Certified

A non-certified supervisor shall have 3 years of reactor experience, including 6 months onsite, prior to assuming independent shift duties.

- (e) Maintenance Personnel

Maintenance and Electrical and Instrumentation (E&I) personnel shall have a year of related craft experience, including 3 months onsite, prior to assuming independent duties on safety systems.

- (f) Plant Technical Support Personnel

Technical support staff shall have 1 year of related reactor engineering experience, including 3 months onsite, prior to assuming independent duties.

(g) Components Handling Reactor Operator

A CH-RO shall have 1 year of reactor experience, including 6 months onsite, prior to assuming independent duties.

(h) Components Handling Senior Reactor Operator

A CH-SRO shall have 3 years of reactor experience, including 6 months onsite, prior to assuming independent duties.

(i) Components Handling Supervisor - Certified

A Supervisor shall have 3 years of reactor experience, including 6 months onsite, prior to assuming independent duties.

(j) Components Handling Supervisor - Non-Certified

A Supervisor shall have 3 years of reactor experience, including 6 months onsite, prior to assuming independent duties.

(4) Training

Contractor operations and support staff training requirements are discussed in section 10.2 of this report.

(5) Reactor Control Manipulations

Reactor control manipulations are required to demonstrate acceptable proficiency in critical control processes above and beyond that required under the training and experience requirements. Use of a plant-specific simulator is an acceptable alternative for performance of training and evaluations of reactor control manipulations which involve startup and significant power changes.

Alternatives to heat generating fuel movements include the use of dummy assemblies, shield plugs, targets or other assemblies manipulated in a manner to simulate fuel movements.

(a) Reactor Building Operator

RBOs do not independently manipulate control or safety rods, adjust primary or secondary coolant flows, or operate other equipment which directly affects reactivity or reactor power during reactor operations or fuel movements. Therefore, this section is not applicable to this position.

(b) Central Control Room Operator

The minimum criteria established for restart shall be two evaluated control manipulations within 6 months prior to restart.

One manipulation shall be an approach to criticality and the other manipulation shall be a power ascension after criticality. These manipulations shall be evaluated by personnel other than his/her immediate supervisor and the evaluation documented.

(c) Supervisor/CCRS-Certified

The minimum criteria established for restart shall be two evaluated control manipulations within 6 months prior to restart. One manipulation shall be an approach to criticality and the other manipulation shall be a power ascension after criticality. Evaluations will be performed of a Supervisor's direction and oversight of control manipulations. Supervisors will not be required to actually perform manipulations. These manipulations shall be evaluated by personnel other than his/her immediate supervisor and the evaluation documented.

(d) Supervisor - Non-Certified

Non-Certified Supervisors do not independently manipulate control or safety rods, adjust primary or heat exchanger cooling water flows, or operate other equipment which directly affects reactivity or reactor power during reactor operations or fuel movements. Therefore, this section is not applicable to this position.

(e) Maintenance Personnel

Maintenance personnel do not manipulate control or safety rods, or adjust primary or heat exchanger cooling water flows. Therefore, this section is not applicable to this position.

(f) Plant Technical Support Personnel

Technical support personnel do not manipulate control or safety rods, or adjust primary or heat exchanger cooling water flows. Therefore, this section is not applicable to this position.

(g) Components Handling Reactor Operator

CH-ROs do not manipulate control or safety rods within the reactor, adjust primary or heat exchanger cooling water flows, or run the Charge/Discharge Machines. Therefore, this section is not applicable to this position.

(h) Components Handling Senior Reactor Operator

CH-SROs shall be required to demonstrate adequate performance of charge and discharge operations using mockup fuel assemblies or other components prior to assuming independent duties if fuel movements have not been performed within the past 6 months. These operations are to be evaluated by personnel other than his/her

immediate supervisor and the evaluation documented.

(i) Components Handling Supervisor - Certified

CH-Supervisors shall be required to demonstrate adequate performance of charge and discharge operations using mockup fuel assemblies or other components prior to assuming independent duties if fuel movements have not been performed within the past 6 months. These operations are to be evaluated by personnel other than his/her immediate supervisor and the evaluation documented.

(j) Components Handling Supervisor - Non-Certified

CH-Supervisors do not manipulate control or safety rods within the reactor, adjust primary or heat exchanger cooling water flows, or operate the Charge/Discharge Machine. Therefore, this section is not applicable to this position.

(6) Examinations

(a) Examinations shall be given to all personnel to validate retention and understanding of material required to complete the tasks of that position. Examinations shall consist of written exams, practical demonstration or drills, oral and operating exams, and simulator exams (where appropriate simulators are available). The exams shall be based on job performance requirements, training conducted, and may include the use of work or reference material routinely available in the job environment such as procedures and/or technical specifications. For certified operators and supervisors, the following criteria apply:

- o Written exams on individual topics and comprehensive exams shall be required. A passing grade of 80% shall be established.
- o Demonstration and/or performance of selected tasks evaluated by personnel other than the immediate supervisor.
- o Oral and Operating exams shall be required. The exam should include plant walkthroughs identifying equipment, procedures, and knowledge required to handle normal and abnormal events.
- o Simulator exams shall be required. Both team performance and individual performance shall be evaluated.

(b) Maintenance personnel currently serving in positions for which formal training and examinations have not been required will continue to function under the existing qualification requirements until such time as training and examinations are developed and implemented.

- (c) In addition to the standard examination process for operations personnel, a management assessment shall be conducted on the knowledge, attitude, and performance of each crew in the simulator and in the actual plant.

(7) Medical Qualifications

Medical qualifications shall be consistent with the motor and mental skills required to perform the job tasks of the position. In addition, personnel who are certified/qualified/assigned to RBO, CCRO, and Supervisory positions shall meet medical requirements specified in ANSI/ANS 3.4, "Medical Certification and Monitoring of Personnel Requiring Operator Licences for Nuclear Power Plants" (Reference 4).

(8) Certification

Certification is a second-level verification that personnel have demonstrated the desired levels in every qualification category (education, experience, medical, training, and job performance) and are periodically evaluated to assure they maintain that level. Certification is required for those positions which independently manipulate or direct the manipulation of reactor controls, fuel charge/discharge machines, or equipment which directly impacts the reactivity of the core. Certification shall be valid for a period of 2 years. Positions requiring certification shall include:

- o Center Section Shift Manager
- o Central Control Room Supervisor/Shift Manager
- o Central Control Room Operator
- o Components Handling Crane Supervisor
- o Components Handling Senior Reactor Operator
- o Other Support Positions as defined in Reactor Department Standing Instructions

(9) Records

- (a) All records involving qualification and certification of individuals shall be developed, completed, and maintained consistent with draft DOE Order 5480.XX. The latest documents that support qualification/certification shall be maintained in an auditable form. Disposition of records 5 years after separation shall be in accordance with DOE Order 1324.2A, "Records Disposition" (Reference 5).
- (b) Personnel need not be required to repeat courses or exams in the event documentation has not been previously retained to requirements unless that documentation is required to support the

current certification or qualification determination (e.g., recertification exams from 1980 are not required to support a 1989 recertification review).

(10) Staffing

- (a) Per the requirements of draft DOE Order 5480.XX, sufficient staff shall be established to operate the reactor in a safe and efficient manner with adequate technical knowledge on shift. Specifically:
 - o Control Room staffing levels shall be established and minimum numbers required for the safe operation of the facility under normal and emergency conditions shall be specified in Technical Specifications. This staffing level shall be based upon historical data, drills, simulator exercises, and supporting analysis.
 - o Staffing levels for the balance of plant shall be established and minimum numbers needed to support the safe operation of the facility under normal and emergency conditions shall be specified in Technical Specifications.
 - o Staffing levels shall be established to provide technical support and maintenance of the reactor facility. Sufficient numbers of personnel shall be on shift or on call to provide required facility support. Call lists for management and technical support personnel shall be available within the Central Control Room (CCR).
 - o Engineering and accident assessment expertise shall be provided on each shift. Such expertise may be met by using a certified supervisor with an engineering degree (STE) who has successfully completed plant specific training, or by training of non-degreed personnel to a comparable technical level.
- (b) The Supervisor in charge of the control room shall be "experienced" (4 years of nuclear power plant experience with at least 6 weeks above 20 percent power as discussed above). A Supervisor shall be in the CCR any time irradiated fuel is in the tank or reactivity manipulations are in progress. While in charge of the control room, the Supervisor shall not leave the control room area.
- (c) A degreed supervisor with STE training shall be on shift and routinely capable of reporting to the control room upon demand anytime the reactor contains heat generating assemblies or when reactivity manipulations are in progress.
- (d) The STE shall be present in the control room during reactor startup and shall report to the control room within 10 minutes in

the event of a scram or off-normal event.

- (e) Three Central Control Room Operators shall be in the control room anytime irradiated fuel or reactivity manipulations are in progress.
- (f) Three Reactor Building Operators shall be within the exclusionary fence anytime irradiated fuel or reactivity manipulations are in progress.
- (g) The Shift Advisor is a temporary coach to the shift crew on administrative practices and does not provide technical advice on operating the reactor. Therefore, no constraints on reactor operations shall be tied to a Shift Advisors presence in the reactor areas. However, every effort should be made to have a Shift Advisor or substitute present with his assigned crew during operating and training periods. A Shift Advisor shall be assigned to each crew for a period of at least 12 weeks after restart.
- (h) Minimum criteria for plant staffing shall be incorporated in Reactor Department Standing Instructions, DPSOP 336, and be approved by DOE prior to restart as a minimum.

10.1.3 Evaluation

Similar to actions taken at commercial plants after the TMI accident to place a Shift Technical Advisor on shift to compensate for operator knowledge deficiencies in engineering principles, SRS has undertaken a program to establish Shift Technical Engineers on shift. The qualifications of these individuals were established in the original Charlotte Criteria, but have been modified by the restart criteria.

The requirements of Charlotte Criteria HP-3, "Technical Support Engineer," have been modified to delete the requirement that this individual be from the Reactor Technology Department. Sufficient time has elapsed to hire degreed engineers into the Reactor Operations Department and certify these individuals as central control room supervisors and qualify them as Shift Technical Engineers. These individuals possess the equivalent academic background and the specialized training outlined in the original criteria. In lieu of the introductory training outlined for Reactor Technology personnel in DPSP 82-1002, these personnel have successfully completed the training required for central control room supervisors and have demonstrated a knowledge of fundamentals equivalent to that of commercial reactor operators. Because these individuals are permanent ROD shift personnel, modifications to ROSI III.1, "Organization of the Reactor Operations Department" (Reference 6), to include the roles and responsibilities of the Shift Technical Engineer have been made instead of creating a separate charter and schedule as required by HP-3.1 and 3.5. These individuals possess equivalent academic credentials, have completed more training than required by the original Charlotte Criteria, and are an integral part of the crew instead of an added resource; therefore, this modification is considered to be an improvement to the original Charlotte Criteria. This longer term improvement was alluded to in the original

recommendation, which states: "The additional support will be provided until operations training in the safety envelope and bases has been satisfactorily implemented."

In addition to the STE position, WSRC has established a temporary shift advisor position. The intent of this position is to provide on-shift coaching to improve the manner in which operations personnel conduct operations, with a goal of promoting professional, safe, and reliable operations consistent with commercial practices. This individual is required to have been licensed by the NRC within the last 5 years and have a minimum of 5 years experience. Contractual arrangements have been established that implement the restart criteria, including that a shift advisor be present on shift for a minimum duration of 5 weeks prior to operations and 12 weeks after criticality. ROSI III.1 discusses the functions and duties of the shift advisor in more detail. The ROSI lists specific areas in which the Shift Advisors are expected to implement improvements, i.e., logkeeping, shift turnover, communications, procedural compliance, work control, and the use of technical specifications.

The highest level of administrative controls that outlines the minimum numbers of personnel required to be on shift and within prescribed duty stations is Section 5 of the Technical Specifications. Section 5, the Administrative Controls section of the Technical Specifications, details the minimum number of operations personnel required to safely operate the reactor facility in the various modes (i.e., startup, level operations, shutdown). Reactor Department Standing Instructions (ROSI) are the implementing procedures that identify both normal and minimum staffing levels and include the non-operations support personnel needed immediately during off-normal circumstances.

ROSI III.5 "105 Building Operating Crews" (Reference 7), specifies the minimum shift crew staffing levels and compensatory actions required if these levels are not met. The ROSI presents both the Technical Specification requirements and the normal shift crew staffing levels for the various modes of operation. ROSI III.5 also includes minimum staffing levels for support personnel, such as the Health Protection, Maintenance, and Power personnel who are required for immediate responses during off-normal conditions.

In addition to specifying the minimum number of personnel required to be on shift, controls are established on where the boundaries of duty stations are and when formal relief is required when exceeding these boundaries. ROSI III.5 defines the control room proper as the area in front of the instrument and control panels, including the shift manager's office. Technical Specifications and ROSI III.5 require a minimum of one certified CCR supervisor in the control room at all times when there is fuel and moderator in the reactor. ROSI III.5 authorizes the control room supervisor to leave the CCR proper only momentarily to take readings and make brief checks behind the instrument panels. These checks are required to be kept as brief as possible and must not include work which would divert attention from CCR operations.

As discussed previously, to compensate for the reduced level of engineering and fundamentals knowledge of the operating crews, an individual qualified as an STE is assigned to each shift to provide engineering and accident

assessment expertise. The Technical Specifications and ROSI III.2 "Supervisory Coverage in the Control Rooms (CCR) Center Section and Components Handling" (Reference 8), require that the STE be available to report to the CCR within 10 minutes following a scram or operational anomaly. During reactor startup the STE is required to be in the CCR to ensure that proper reactivity parameters have been correctly designed and incorporated into the reactor charge and that the core is behaving as projected.

The Technical Specifications and ROSI III.5 require that at least two certified CCR operators be present in the CCR at all times when the reactor is shutdown and the tank contains fuel and moderator. During reloading of the reactor, the ROSI requires four CCR operators be present in the CCR at all times, although the Technical Specifications require only three. For reactor startup, ROSI III.5 requires seven CCR operators to be on shift, and both the Technical Specifications and ROSI require five to be in the CCR. During power ascension, level power, and control rod movement from the trim panel, the ROSI requires at least four CCR operators to be on shift, and both the Technical Specifications and ROSI require three in the CCR at all times. Additionally, ROSI III.5 requires four components handling Senior Reactor Operators to be in the crane control room during charge and/or discharge operations of fuel and target assemblies.

The Technical Specifications and ROSI III.5 require that at least two reactor building operators be part of the shift crew when the reactor is shutdown and the tank contains unirradiated fuel and moderator. For startup and operations, four reactor building operators are required to be part of the shift crew by technical specifications. Additionally, ROSI III.5 requires four components handling reactor building operators cover assembly/disassembly areas during charge and/or discharge operations of fuel and target assemblies.

In addition to operations personnel, ROSI III.5 requires a minimum level of support staffing. As a minimum, one Electrical and Instrument (E&I) technician, one health protection supervisor, three health protection inspectors, and two power operators shall be on each shift. Although staffing levels are established to provide the anticipated support and maintenance needs of the reactor facility, control room personnel must have an effective means of contacting management and other support personnel during unexpected circumstances. A call list has been developed to provide the on shift crew ready access to management and technical support when needed. In addition, emergency notification lists have been established for reportable and emergency notification events.

In addition to revising staffing levels on shift, WSRC has also revised the qualification criteria for key positions. DPSOP 38, "Reactor Personnel Selection, Qualification, and Training Manual" (Reference 9) identifies the education, experience, and training levels for personnel who routinely work in the reactor facilities. This includes personnel in the Reactor Operations and Maintenance, Reactor Training and Procedures, Reactor Engineering, and Health Protection departments. All position qualifications identified in the restart criteria are met by DPSOP 38. Table 10.1-1 is a summary comparison of the qualifications. DPSOP 38 and the restart criteria are based on ANSI/ANS 3.1. As required by DOE Orders, DPSOP 38 identifies any deviation from the ANS

standards and provides a written justification for that deviation. DPSOP 38 has been reviewed and approved by the DOE.

DPSOP 38 specifies the minimum training, examination, evaluation, and practical experience requirements for achieving and maintaining certification, recertification, qualification, and requalification. It provides definitions of the terms certification, medical evaluation, and qualification and lists the plant positions requiring qualification and certification. Included in this list are the following operations positions: (1) Reactor Building Operator; (2) Central Control Room Operator; (3) Building Supervisor; (4) Central Control Room Supervisor; (5) Shift Manager; (6) Components Handling Reactor Operator; (7) Components Handling Senior Reactor Operator; (8) Assembly and Disassembly Supervisor; (9) Charge and Discharge Machines Supervisor; and (10) Shift Technical Engineer. It also provides the detailed requirements for qualification, certification, and recertification programs for reactor operations personnel. DPSOP 38 outlines the course content of the training programs, the required simulator training, on-the-job training, and special requirements required to keep the certification current, such as number of shift watch requirements per quarter.

DPSOP 38 also identifies the training records that should be considered as permanent records and must be retrievable. This list includes things such as completed tasks and written examinations. Official records are to be filed and a copy of completed training records sent to the Plant Records Department for microfilming and storage. The Reactor Operations Procedures Index contains the specific retention requirements for almost all operating records including the individual certification and qualification records. All records used for certification and qualification have a specified retention period of the lifetime of the plant. In addition, the Training Administration Manual (TAM) outlines more detailed requirements for how training records are to be handled.

DPSOP 38 requires that personnel assigned to CCR Operator or Supervisor positions must meet the medical requirements specified in ANSI/ANS 3.4. In addition, the fitness for duty criteria are also outlined, including pre-employment and random drug testing.

DPSOP 38 requires CCR operators and supervisors perform two control manipulations within 6 months prior to restart to demonstrate proficiency at handling the controls after an extended period of inactivity due to the reactor being shutdown. One control manipulation is to be an approach to criticality and the other is a power ascension after criticality. The CCR supervisor will only be evaluated on his/her direction and oversight of control manipulation; while the control room operator will be evaluated on his/her actual performance of control manipulations.

DPSOP 38 requires Components Handling Senior Reactor Operators and the certified Components Handling Supervisors to have classroom and on-the-job training for the C&D machine systems and their operation. Maintaining an active certification requires that these personnel be assigned duties in components handling at least one work shift per calendar quarter. Components Handling Senior Reactor Operators and Charge and Discharge Machine Supervisors

are required to perform a sufficient number of control manipulations to maintain and demonstrate adequate proficiency performing charge and discharge operations prior to fuel reloading if fuel movements have not been performed within the past 6 months.

In addition, the DOE staff reviewed the WSRC closure packages associated with staffing and qualification and determined that all items had been adequately addressed and no outstanding issues remained open.

10.1.4 Conclusion

The staffing and qualification program review is now complete. The DOE Staff has determined that the current program delineating the requirements for plant staffing and qualifications meets the requirements of the restart criteria specified in the SER.

10.1.5 References

1. American Nuclear Standards Institute/American Nuclear Society, ANSI/ANS 3.1-1987, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants." 1987.
2. U. S. Department of Energy, DOE Order 5480.6, "Safety of DOE- Owned Reactors."
3. _____, DOE Order 5480.xx, "Personnel Selection, Qualification, and Training Requirements."
4. American Nuclear Standards Institute/American Nuclear Society, ANSI/ANS 3.4, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants."
5. U. S. Department of Energy, DOE Order 1324.2A, "Records Disposition."
6. WSRC, ROSI III.1, "Organization of the Reactor Operations Department," Revision 5, July 1990.
7. WSRC, ROSI III.5, "105 Building Operating Crews," Revision 8, September 1990.
8. WSRC, ROSI III.2, "Supervisory Coverage in the Control Rooms (CCR) Center Section and Components Handling," Revision 5, August 1990.
9. WSRC, DPSOP 38, "Reactor Personnel Selection, Qualification, and Training Manual," Revision 35, October 1990.

TABLE 10.1-1

SER and DPSOL 38 Comparison

Position	SER Minimum Education Requirements	DPSOP 38 Minimum Education Requirements	SER Minimum Experience Requirements	DPSOP 38 Minimum Experience Requirements
Reactor Building Operator (RBO) (Non-certified)	High School diploma or equivalent	High School diploma or equivalent	1 year reactor experience, including 6 months onsite	1 year nuclear and 6 months onsite
Central Control Room Operator (CCRO) (Certified)	High School diploma or equivalent	High School diploma or equivalent	3 years reactor experience including 6 months onsite	3 years at production reactor including 6 months as qualified RO and 6 months onsite
Supervisor/Central Control Room Supervisor (Certified)	High School diploma or equivalent	High School diploma or equivalent	3 years reactor experience including 6 months onsite. Additionally, control room supervisors shall have 6 weeks experience at an operating SRS reactor at >20% power	3* years nuclear plant experience including 1 year as certified SRO - Center Section or BS and 1 year of reactor experience and 6 weeks >20% power
Support Staff Supervisor (Non-certified)	High School diploma or equivalent	High School diploma or equivalent	3 years reactor experience including 6 months onsite	3 years nuclear plant experience including 1 year as certified SRO - Center Section or BS and 1 year of reactor experience and 6 months onsite
Maintenance Personnel (Mechanical & Electrical)	High School diploma or equivalent	High School diploma or equivalent	1 year related craft experience including 3 months onsite prior to working on safety systems	2 years in craft, including one year at a nuclear facility and 6 months onsite

* 4 years required for restart, and at least one supervisor on shift must be STE qualified (BS, certified supervisor, completed STE training)

TABLE 10.1-1 (Cont.)

Position	SER Minimum Education Requirements	DPSOP 38 Minimum Education Requirements	SER Minimum Experience Requirements	DPSOP 38 Minimum Experience Requirements
Plant Technical Support Engineering	BS degree in engineering or related	Bachelor's degree in engineering or related	1 year related reactor engineering experience,	1 year nuclear plant experience
Components Handling Reactor Operator (Non-certified)	High School diploma or equivalent	High School diploma or equivalent	1 year reactor experience including 6 months onsite	1 year nuclear including 5 months in assembly/disassembly and 6 months onsite
Components Handling Senior Reactor Operator (Certified)	High School diploma or equivalent	High School diploma or equivalent	3 years reactor experience including 6 months	3 years nuclear plant experience including 6 months onsite
Components Handling Supervisor (Certified) (C&D Machine)	High School diploma or equivalent	High School diploma or equivalent	3 years reactor experience including 6 months onsite	3 years nuclear plant experience including 1 year as certified SRO - Center Section or BS and 1 year of reactor experience and 6 months onsite
Components Handling Supervisor (Non-certified) (Assembly/Disassembly)	High School diploma or equivalent	High School diploma or equivalent	3 years reactor experience including 6 months onsite	3 years nuclear plant experience including 1 year as certified SRO - Center Section or BS and 1 year of reactor experience and 6 months onsite

10.2 Training

10.2.1 Background

Training and qualification programs are designed to assure that only those individuals capable of safely performing their assigned duties are authorized to operate the reactor facilities. A good training program is necessary in order to achieve a high level of personnel performance and professionalism and to meet the high operational standards required for plant safety and reliability. Training programs must be sufficiently broad in scope to ensure that the knowledge necessary for safe plant operation is systematically reviewed and is flexible enough to address recent operating experiences, identified performance problems, plant modifications, and procedure changes in a timely manner.

Operator and support staff training is a fundamental reactor safety issue at SRS. One of the major issues contributing to the prolonged shutdown of the K-Reactor following its routine refueling outage in April 1988 was the performance of personnel during the conduct of operations and support duties. After the Three Mile Island (TMI) accident in 1979, significant attention and resources were devoted to upgrading the technical qualifications of operators at commercial nuclear facilities. Although increased attention was also focused on SRS training programs, the rate and effectiveness of these changes did not keep pace with those in the commercial sector.

The fact that SRS training does not meet industry standards is a driving force behind the new push for excellence. The need for and desire to obtain increased operator knowledge levels is recognized and strongly endorsed by WSRC management. This fact, alone, represents a significant cultural change in the manner in which the reactors are operated and training conducted.

Now that WSRC operations and plant management acknowledges and supports the need for training improvements and insists that the upgrades begin near term, the process of training can improve. Staffing and management changes in the training department have resulted in significantly increased awareness of commercial and naval training standards and practices. Numerous short- and long-term programs have been implemented to address the weaknesses in operator knowledge; however training of this magnitude is not a short-term solution, and compensating measures are necessary until long-term programs become effective.

Specific areas of weakness in the training programs at SRS were highlighted for completion prior to restart as part of the Charlotte Criteria, the and WISR evaluations, and DOE internal and external reviews. Curricula have been defined for operations personnel, Shift Technical Engineers, components handling personnel, management, maintenance, and other support personnel.

10.2.2 Restart Criteria

The criteria for determining the adequacy of the SRS training program for restart are described below.

(1) General

The SRS restart training program shall be performance-based and shall include the following essential elements:

- (a) systematic analyses of job functions and tasks
- (b) documented learning objectives that are required for successful completion of job functions and tasks
- (c) effective training program design and implementation of the learning objectives
- (d) evaluations of trainee mastery of the learning objectives
- (e) evaluation and revision of training programs based upon trainee evaluations and job performance

Restart training shall be conducted by (or under the direction of) qualified instructors. Materials and training programs for restart training shall be developed by qualified experts and approved by SRS Operations and Training Department managers.

(2) Operations Personnel Training

In addition to meeting all training requirements for initial certification and for recertification, the restart training shall address the following:

- (a) major plant modifications that have occurred during shutdown
- (b) significant operating, emergency, or administrative procedure changes resulting from plant modifications or changes in operating philosophy
- (c) teamwork and diagnostic skills
- (d) reactor engineering fundamentals specific to the SRS reactor design and design modifications
- (e) communications
- (f) simulator training, including:
 - o Plant Modifications Training -- Training shall be provided that covers hardware changes made to the reactor, including seismic modifications, the moderator recovery system or MRS, the fourth emergency cooling water system addition, the new tritium monitors, the lube oil system for ac and dc motors, and the leak detection system. Operating crews shall receive training in the classroom and, when possible, on the simulator in procedures associated with these modifications.

- o Procedure Training -- Training shall be provided that covers operating and emergency procedures developed or modified to support major plant modifications, including Technical Specifications, Technical Standards, and the Technical Specification Linking Document; administrative procedures modified to support changes in operating philosophy and conduct of operations such as logkeeping, notification requirements, turnover, equipment tagouts/lockouts, etc.; and the new emergency action level procedures. For symptom-based emergency procedures, classroom and simulator training shall emphasize teamwork and communications as well as diagnostic skills. Symptom-based simulator training scenarios shall challenge operators to use new, flow-diagram procedures effectively.
- o Teamwork/Diagnostic Skills Training -- Training shall be provided on teamwork and diagnostic skills. This training shall be specifically addressed and integrated into all classroom, on-the-job, and simulator training modules.
- o Fundamentals Training -- Fundamentals training shall be conducted prior to restart and shall concentrate on the areas identified in operating experience reviews (e.g., reactor theory, reactivity assessment, applied pump and valve theory, applied process instrumentation). In addition to the technical aspects and mechanics of the fundamentals topics, training shall be provided on the reasons for actions being taken.
- o Communications Training -- Training shall emphasize a disciplined formal approach to reactor operations and communications having the characteristics of inquisitiveness, attentiveness, preparedness, and accountability, described as the "new professionalism." In conducting communications training, it is recommended that role playing scenarios be developed to enhance realism. Self-critiques should be utilized to encourage team building. Operational functions to be stressed should include, but not necessarily be limited to: logkeeping; equipment tagouts/lockouts; work permit controls; shift turnover; plant configuration control; and procedure compliance. Development of communication training scenarios shall also consider:
 - lessons learned from SRS reactor unusual occurrences and from the commercial nuclear industry
 - responses to anomalies and off-normal events
 - roles and responsibilities of oversight groups

- changes in administrative controls
- reporting thresholds

(3) Shift Technical Engineer

WSRC shall have a Shift Technical Engineer (STE) on each reactor crew team or shift. The STEs shall successfully complete training in the following areas:

- (a) safety envelope, technical specifications, and technical specifications linking document
- (b) startup operations
- (c) authorities and responsibilities
- (d) charge/discharge operations
- (e) core physics/thermal hydraulics
- (f) safety analysis
- (g) lessons learned
- (h) operational engineering calculations including heat balances, critical rod positions, and safety limit verifications

(4) Other Operations and Support Personnel Training

Other reactor operations and support personnel shall be trained prior to restart, with concentration on changes in the conduct of operations and on specific job-related training. Training shall be provided for maintenance and support personnel staff including management and supervisory staff, as well as for personnel in health protection, electrical, instrumentation and technical support areas.

- (a) Maintenance Personnel -- See Section 13 for details on maintenance personnel training requirements.
- (b) Radiation Protection/Health Protection -- See Section 4 for radiation/health protection personnel training requirements.
- (c) Component Handlers -- Training prior to restart shall include communications, reporting, teamwork, and diagnostic skills; changes in the conduct of operations; applicable hardware/procedure modifications; and significant operating experience events. After more than 6 months inactivity, drills shall be conducted prior to any fuel movement to demonstrate proficiency.

- (d) Management -- Training for management shall include formality of operations; detection and reporting of anomalies; reactivity assessment; emergency action levels; lessons learned from SRS and commercial industry reactors; overview of other training; the restart plan; reporting levels and thresholds; safety envelope; administrative controls; and basis for cultural change. This training shall focus on those aspects of plant hardware/procedure modifications, work practices, etc. in which management is either directly involved, or for which management must set the example for plant personnel.
- (e) Supervisory -- Supervisory personnel shall be trained in the basis for culture change and restart plan; formality of operations; communications; teamwork; procedure compliance; control of combustibles; housekeeping; changes in work practices/procedures; nonconformance reporting; and changes in administrative controls.
- (f) WSRC Reactor Safety Evaluation Division -- Personnel in the WSRC Reactor Safety Evaluation Division shall receive training that specifically includes roles and authorities; organizational relationships; INPO observation training; reporting levels and thresholds; system descriptions; safety envelope; administrative controls; restart plan; and basis for cultural changes.

(5) Operator Evaluation Prior to Restart

- (a) Each of the operator training programs for SRS shall include defined, testable training objectives. Trainee performance shall be evaluated on the basis of written tests, simulator performance, and in-plant demonstrated capabilities. Testing methods and procedures shall ensure that testing objectives are satisfied. In addition, operator training records shall be maintained in an auditable form. Operator retraining in operating skills required for control manipulation is required prior to restart in view of the extended shutdown of the SRS reactors.
- (b) Every senior reactor operator at SRS shall perform at least two evaluated control manipulations on the simulator prior to restart, including one critical approach and one power escalation. SRS supervisors shall be evaluated on ability to direct controlled manipulations in the simulator prior to restart.

(6) Management Assessment

WSRC shall perform a management assessment of the operator training program during restart training. The assessment shall be made on the basis of: (1) written and oral examinations; (2) simulator performance evaluations; and (3) walk-through evaluations under operating conditions. WSRC senior management shall observe crew performance during simulator scenarios.

10.2.3 Evaluation

The requirements of Charlotte Criteria HP-3, "Technical Support Engineer," have been modified to delete the requirement that technical support engineers be from the Reactor Technology Department. Sufficient time has elapsed to hire degreed engineers into the Reactor Operations Department (ROD), certify them as central control room supervisors, and qualify them as Shift Technical Engineers. These individuals possess the equivalent academic background and the specialized training outlined in the original criteria. In lieu of the introductory training outlined for Reactor Technology personnel in DPSP 82-1002, these personnel have successfully completed the training required for central control room supervisors and have demonstrated a knowledge of fundamentals equivalent to that required of commercial reactor operators. Because these individuals are permanent ROD shift personnel, ROSI III.1, "Organization of Reactor Operations" (Reference 1), has been modified to include the roles and responsibilities of the Shift Technical Engineer instead of creating a separate charter and schedule as required by HP-3.1 and 3.5. These individuals possess equivalent academic credentials, have completed more training than what is required by the Charlotte Criteria, and are an integral part of the crew instead of an added resource. The staff, therefore, considers this modification to be an improvement to the original Charlotte Criteria. This longer term improvement was alluded to in the original recommendation, which states: "The additional support will be provided until operations training in the safety envelope and bases has been satisfactorily implemented."

Training Administration

DPSOP 38, "Reactor Personnel Selection, Qualification, and Training Manual" (Reference 2), provides minimum compliance requirements for meeting DOE Orders and commercial industry training standards. The Training Administration Manual (TAM) (Reference 3) is the administrative control document that details the manner in which training programs are to be developed, presented, and evaluated. The TAM contains the four-step process used as a systematic method for ensuring that the tasks required for competent job performance are identified, documented, and included in the appropriate training programs. The four steps to be performed are a needs assessment, a job analysis, a task selection, and a task analysis.

For many of the restart training topics, the initiator, or assessment of need, was the Charlotte Criteria/WISR and other internal and external reviews. Due to the nature of the restart training topics, formal job and task analyses (JTAs) are not required to determine the important elements for training. However, a structured development process is required to assure all important elements are addressed. A systematic method for developing training material, such as a table-top functional analysis, is an acceptable alternative to the formal JTA.

Although not well documented, WSRC did use a structured development process for development of the restart training material. This development process centered around a table-top functional analysis performed by WSRC training personnel, training subcontractors, and DOE staff personnel. After development

of the necessary training material, this material was reviewed and approved by management personnel from the Reactor Training and Procedures organization, Reactor Operations Department, and DOE training personnel.

The TAM describes the methodology for constructing terminal learning objectives. The intent of determining learning objectives is to describe what the student should be able to do after being trained in this area. The training materials reviewed contained terminal learning objectives that the staff determined were appropriate to the material being taught. The TAM also discusses the construction of enabling objectives for each skill and knowledge requirement for successful performance of a desired task. These enabling learning objectives support achievement of the associated terminal learning objective. An enabling learning objective is written for each skill and knowledge requirement necessary for successful completion of each task. The training materials reviewed contain enabling objectives that the staff determined were appropriate to the material being taught and supportive of the terminal objective specified for that material.

The TAM describes the final approval process for training programs and materials. It specifies the approvals required for training materials prior to use, including approval by management of both the Operations and the Training departments.

The TAM contains guidance on examination preparation, review, and approval to properly measure a student's mastery of skills and knowledge (e.g., principles, concepts, rules, etc.) before, during, and after instruction. The TAM describes the purpose and composition of written components of the examination process. It requires the student to write down responses to recognition-type test items such as multiple choice, matching, and true-false questions and recall-type test items such as completion, short answer, and essay questions. The TAM requires that one or more written examination items be constructed to measure student achievement of the performance requirements specified in each enabling learning objective. Appropriate types of test items are selected and written based upon the classification of the enabling learning objectives. DPSOP 38 specifies the pass/fail criteria for certification/qualification and requires a passing score of 80 percent or better for successful course completion. The DPSOP requires that a final, comprehensive examination, which measures student achievement of the learning objectives, be administered at the end of every formally scheduled training course. Written exams are routinely administered at the completion of each week of cycle training.

Examination results are reviewed by examination graders, instructors, and training management. The reviews are intended to identify potentially invalid test questions, areas of instructional weakness, or student knowledge weakness. Automatic re-grading of borderline grades and failures is required to be performed by a grader other than the original examination grader. In addition to examination reviews for test validity, operations management uses the results of examinations when establishing the crews for K-Reactor.

The TAM describes the purpose and composition of the simulator examination process. This process uses job performance measures as one component of the evaluation and is used for those situations in which the student is asked to apply training, knowledge, and psychomotor skills to the performance of actual or simulated tasks. OPS-ROTP-890660, "Policy on Performance Evaluation and Testing Using the Simulator" (Reference 4), states that two areas of performance should be checked in the simulator: individual performance and team performance. Simulator training has been integrated into each week of cycle training for application of new topics and reinforcement of previous training. Formal simulator evaluations are required as part of the management assessment program, peer evaluator program, and control manipulation exercises in addition to the weekly simulator exercises.

The TAM describes the purpose and composition of the oral component of the examination process. This component is used when it is desirable for the proctor to have flexibility in probing problem areas and to allow the student to express responses and concepts in his own words. The oral examination may be performed as part of simulator exercises, in-plant drills, or as an independent examination. Two-day oral and operating examinations are required as part of initial certification, and 1-day oral and operating examinations are required for recertification. In addition, after successful completion of the Peer Evaluation, an operator is required to pass a collegiate-style oral board chaired by the Plant Manager. Likewise, supervisors are required to complete an oral board chaired by the Site President.

The TAM describes the process of conducting on-the-job training (OJT). OJT is a systematic method of transferring the required job-related skills to the student by providing the opportunity for the student to practice the actual skills needed in the job environment. OJT is used to reinforce lessons that have been learned in the classroom and to help the students gain a knowledge of component location, function, and method of operation. ROSI V.7, "Onshift Training Program" (Reference 5), describes the mechanics of the OJT program. The ROSI outlines the various program elements required for an effective OJT program. These include identification of the operator job functions requiring OJT, development of the training materials required for those job functions, establishing performance standards and evaluating operators performance against those standards, and ensuring documentation of this process. Roughly 20 OJT system qualification cards are scheduled for completion prior to restart.

The TAM discusses the methods to be used to evaluate and revise training courses based on instructor evaluations, student course evaluations, examination results, post-training performance evaluations, and verbal feedback.

DPSOP 38 specifies the minimum education and experience requirements training instructors need in order to comply with DOE Order 5480.6, "Safety of DOE-Owned Reactors" (Reference 6), and to be consistent with commercial standards as defined in American Nuclear Society (ANS) Standard 3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants" (Reference 7). The TAM is the actual procedural implementation of the training staff selection, training, and qualification requirements. It

specifies the requirements to be met for personnel developing or revising training material. The requirements meet both the commercial and restart criteria standards for ensuring that personnel who provide instruction are qualified in the subject areas.

The TAM specifies the responsibilities for records retention and authorized disposition. It contains the audit requirements for training records, including an effective access control system, availability and proper location, and documentation of audit results.

Operator Training

The SRS operator training programs were designed to modify the manner in which operations are conducted and to enhance technical levels in selected areas prior to restart. Additional technical enhancement will be made after restart to meet the training levels required by commercial reactor standards. In the interim, compensatory measures such as placing an STE on shift will be implemented. Although much of the training has been conducted, conclusions on the effectiveness of the training will not be established until after the peer evaluator, management assessment, Operational Readiness Review (ORR) evaluations, and implementation inspections have been completed.

In addition to normal recertification requirements, the restart training program for operations personnel includes training on major plant modifications, procedures, changes to administrative controls, teamwork and diagnostic skills, applied fundamentals, communications, emergency response actions, technical specifications, and many other topics. A list of the major subjects taught each cycle is contained in Attachments 1 and 2.

Training conducted on plant modifications that have occurred during shutdown included the following subjects: seismic modifications, addition of the moderator recovery system, fourth emergency cooling system addition point, new tritium monitors, new lube oil system for the AC and DC motors, and leak detection system. Training was also conducted on the operating and emergency procedures associated with the plant modifications. This training consisted of a combination of classroom, simulator, and on-the-job training.

Due to the significant changes being made to the Technical Specifications and related documents, training on the Technical Specifications, Technical Standards, and the Technical Specification linking document was conducted as part of the restart training program. Several new or significantly modified technical specifications were identified early as high priority items, and the development work was completed. This allowed training to be conducted early in the restart program. Later, it was decided to reformat the technical specifications prior to restart. The staff reviewed the training materials associated with Technical Specification training and found them to be acceptable.

Due to recent changes in operating philosophy and conduct of operations (i.e. logkeeping, turnover, notification requirements, equipment control, and tagouts/lockouts), many of the administrative control procedures were revised to support the changes. Training on the changes was conducted as the new

programs were defined and implemented. Subsequent changes have been made and continue to be made to these procedures. To determine if formal retraining or informal training such as shift briefings or required reading are adequate, each of the changes is evaluated. In addition, the combined magnitude of the individual changes is reviewed to determine the degree of training required.

New emergency action level procedures have been developed as part of the emergency response program upgrades. Classroom, simulator, and on-the-job training has been conducted on emergency action levels and has been integrated into all off-normal simulator exercises conducted for continuing emphasis.

Specific classroom training and simulator training dealing with teamwork, diagnostic skills, and communication has been and continues to be conducted as part of cycle training. All three areas are integrated into the soft skills required during all simulator training.

The applied fundamentals training that has been conducted concentrated on the areas identified in operating experience reviews (e.g., reactor theory, reactivity assessment, applied pump and valve theory, and applied process instrumentation) as requiring increased attention as the direct result of plant incidents. The training addresses both the technical aspects of the topics and the reasons for the actions being taken. In addition, reactor building operators have received a significant amount of additional classroom training in basic theory fundamentals (conducted during periods when Central Control Room operators were in the simulator) and other specialized training topics that were outside the RBO job requirements.

DPSOP 38 requires certified Control Room Operators to perform two control manipulations within 6 months prior to restart as specified in the restart criteria. One of these manipulations is an approach to criticality; the other is a power ascension after criticality. These manipulations must be evaluated and documented by personnel other than the immediate supervisor. The purpose of these manipulations is to demonstrate operator proficiency at control rod movements after a prolonged period of inactivity due to shutdown conditions.

DPSOP 38 requires certified Control Room Supervisors to direct and oversee two control manipulations within 6 months prior to restart. One of these manipulations is an approach to criticality; the other is a power ascension after criticality. The supervisor's direction and oversight of these manipulations must be evaluated and documented by personnel other than the immediate supervisor. The intent of this requirement is the same as that for the Control Room Operator. DOE will monitor performance of control manipulations and confirm successful completion of all required manipulations prior to restart. Pending completion of the above, control manipulations remains an open issue.

The staff determined that all operations training includes adequate training objectives, lesson plans, and study guides and that the mechanics of training presentations have been adequate. Deficiencies identified during the course of training were corrected prior to or at the time of material presentation.

Shift Technical Engineer Training

A formal training program has been established for the Shift Technical Engineers (STEs). This program is designed to assure that the STEs are sufficiently knowledgeable to adequately analyze plant conditions in normal and accident conditions and provide the shift manager with an overall assessment regarding the technical specifications and safety analysis report and the general protection of the safety and health of the public, environment, and onsite work force. Prior to entry into the STE training program, individuals must have completed certification training for central control room supervisor, must possess a Bachelor of Science degree in engineering or a related science, and must be selected for overall ability by operations management. The training program consists of classroom and simulator training as outlined in Attachment 3.

In addition to instruction on the theoretical concepts, the STE training program contains practical exercises using these theoretical concepts. These exercises consist of things such as using the procedure to estimate critical rod position during startup or analyzing vital safety parameters during abnormal events.

The STEs are also required to demonstrate a level of knowledge in fundamentals equivalent to that required for commercial programs based on the Institute of Nuclear Power Operations (INPO) recommendations for Senior Reactor Operators. This program consists of classroom instruction, self-study, and testout. Testout was used for a large portion of the fundamentals training because STEs possess Bachelor of Science degrees in engineering or related sciences, such as physics.

Components Handling Training

Components handling personnel are responsible for the movement of fuel assemblies prior to and after irradiation. They also perform the actual loading and unloading (charge and discharge) of the fuel in the reactor. The training program established for component handlers consists of "soft" skills (communications, teamwork, diagnostic skills), changes in the conduct of operations (logkeeping, notifications, housekeeping), and significant operating experience events (both internal and external).

DPSOP 38 requires components handling personnel to perform component handling manipulations within 6 months prior to restart as specified in the restart criteria. The purpose of these manipulations is to demonstrate proficiency at performing these manipulations following long periods of inactivity due to shutdown conditions. These manipulations shall involve actual movement of non-fuel assemblies in a manner similar to actual fuel movements.

The staff determined that all components handling training includes adequate training objectives, lesson plans, and study guides and that the mechanics of training presentations have been adequate. Deficiencies identified during the

course of training were corrected at the time of material presentation. Conclusions on the effectiveness of the training will not be established until after the management assessment, ORR evaluations, and implementation inspections have been completed.

Management Training

The training program established for management personnel is directed at formality of operations. The focus of this training is on management's role in establishing formality of operations as a part of SRS's new professionalism.

The management training program includes an outline of management's responsibility in detecting and reporting anomalies (including a discussion on notification requirements). Training also includes information on reactivity assessments and the procedures associated with reactor control rod positioning. In addition, the program includes training on emergency action levels (EALs) and the expected response to the various levels. Ten categories of abnormal events are defined by DPSOL 100-4000 PLK, "Emergency Classification, Notification, and Initial Actions" (Reference 8), including radiological releases, loss of power, security, etc. Within each of these categories, any one of four EALs can occur (i.e., unusual event, alert, site emergency, general emergency). The training also concentrated on reporting levels and thresholds for less than emergency events.

Part of the management training consisted of a review of lessons learned from SRS and commercial industry reactors. Five events were reviewed and discussed in the study guide. These reactor events were the Browns Ferry fire, the TMI-2 loss-of-coolant accident, the Chernobyl criticality accident, the P-Reactor reactivity event, and the K-Reactor cooling water event.

Management personnel also received an overview in subject areas such as fire protection, leak-before-break, seismic, and communications. The training overviewed the reactor restart plan and items such as plant modifications and upgrades, changes in Technical Specifications, changes in operating procedures, and the purpose of the linking document.

The staff determined that all management training includes adequate training objectives, lesson plans, and study guides. The mechanics of training presentations were also adequate. Deficiencies identified during the course of training were corrected at the time of material presentation.

Supervisory Training

The purpose of the SRS supervisory training program is to train first- and second-level supervisors in support groups such as power, maintenance, and health protection on the new standards of formality of operations, the new professionalism, communications, procedure compliance, and teamwork. While the basis for cultural change is not stated anywhere in the lesson plan material, all sections of the student study guide deal with program changes and requirements being implemented to enhance safety prior to reactor restart.

The staff also determined that the supervisory training program includes the soft skills and communications training includes the principles of closed-loop communications and repeat back of instructions. Training on procedure compliance and common errors associated with use of procedures and blind procedure compliance were highlighted verbatim. Procedure compliance was also identified as one of the examples of increased formality of operations.

Work practices, such as housekeeping and control of combustibles, were also addressed in the training program. Special emphasis was placed on controlling flammable materials in areas where components and systems essential to nuclear safety are located. The training material identified the new programs and controls used to control combustible materials. The training on housekeeping explains how the new housekeeping program will supplement the SRS safety program and how good housekeeping practices can be closely related to safety excellence.

A portion of the supervisory training program was dedicated to changes in work practices/procedures. A discussion on the procedure change program was included in the training material. This discussion focused on the use of Temporary Procedure Changes (TPCs) and when they should be used and Procedure Revision Requests (PRRs) and their applicability.

The supervisory training program did not contain material on nonconformance reporting (NCRs). The staff has determined that this does not fully meet the restart criteria in that there is no discussion within the study guide or lesson plan that covers the subject of nonconformance reporting. Pending completion of training on NCRs, this issue remains an open item.

The staff determined that the supervisory training program includes adequate training objectives, lesson plans, and study guides and that the mechanics of training presentations have been adequate. Deficiencies identified during the course of training (other than the lack of NCR training) were corrected at the time of material presentation.

Reactor Safety Evaluation Section

A training program has been established for WSRC Reactor Safety Evaluation Section (RSES) personnel. The RSES training program requirements have been changed due to changes in the Reactor Operations Management Plan (ROMP). The ROMP implies that the RSES charter as a safety oversight group is completely separate from operations technical support; therefore, the strict training program acceptance criteria requirements of HP-5 no longer apply to the RSES training program. Under the revised requirements, the RSES staff is required to attend and participate in formal classroom training with appropriate text book material, but they are not evaluated as part of that formal training process. Instead, RSES personnel training and qualifications are documented in accordance with RSES 109, "Required RSES Employee Training and Technical Staff Qualification" (Reference 9).

Training records for RSES personnel training are required to be maintained. RSES 109 contains requirements for maintenance of documentation associated with personnel qualification. These training and qualification records are

Quality Assurance records and are maintained and distributed as necessary by the Training/Education Coordinators for the RSES.

RSES training on roles and authorizations, organizational relationships, and administrative controls is taught as part of RSES orientation, which is given to all RSES personnel. Other topics taught as part of this orientation include training on the Safety Department organizational charts, Safety Department charters, Nuclear Reactor Safety procedures that implement reactor safety and technical staff training and qualification requirements, and INPO Observation Training.

As part of RSES personnel training, training on reporting levels and thresholds is required. Part of this training consists of instruction on Emergency Classification Levels (ECLs). RSES personnel also receive system description training. The system description short course is part of a maintenance training course that serves as an introductory course for RSES resident engineers. The course provides an overview on the purpose and function of all safety-related systems in SRS production reactors. Training on the safety envelope is also required for RSES division personnel. In order to satisfy this requirement, RSES staff attended part of Reactor Operations cycle training dealing with old Technical Specifications, new Technical Specifications, Technical Standards, and the linking document.

The RSES Review Techniques Manual training is a required part of the RSES personnel training. This training manual is an in-house training document used to provide guidance and recommended techniques for accomplishing the safety evaluation reviews which RSES performs. The document also addresses related activities such as preparation for reviews and documentation of findings. A good portion of this manual, along with the RSES Section Manual, addresses Administrative Control procedures. While the guidance contained in this manual applies to most reviews and investigations, RSES staff members complete additional training on review and investigative techniques in root cause analysis, MORT A/I, and INPO Evaluation Training.

Training for RSES personnel also includes training on the basis for cultural change and the restart plan. This is a site-wide training course taught to all employees who are now employees of Westinghouse. This training is not applicable to new RSES technical staff members who were hired from outside SRS. Course subject areas related to cultural change include formality of operations and the new professionalism.

Management Assessment Program

The management assessment concept, as originally outlined in WISR 31, "Management Assessment Plan" (Reference 10), was to require active senior management participation in the evaluation of operator performance and readiness to restart. This evaluation would primarily be focused on the "soft" skills such as communications, leadership, teamwork, and diagnostic skills demonstrated by the operator and supervisors during simulator and in-plant exercises. As part of these assessments, technically qualified

individuals would perform the evaluations on hard skills, such as control board awareness and response, procedure compliance, and physical manipulations of equipment controls.

WISR 31 contains guidance on the purpose and conduct of the management assessment process, including the use of simulator exercise evaluations and in-plant evaluations under operating conditions, as a part of the overall assessment of operator readiness to restart. In accordance with the Management Assessment Plan, simulator and in-plant drills have been conducted by WSRC management and other qualified personnel. In addition, operator interviews were conducted, as deemed appropriate, in support of the evaluations. These evaluations and interviews serve as a basis for part of the overall assessment of operator readiness to restart and are documented in interim reports.

Peer Evaluator Program

Each person who will stand watch at K-Reactor after restart will be required to successfully complete the peer evaluation prior to standing watch. The peer evaluator program was established to provide a comprehensive evaluation of operator and crew readiness to safely restart the reactors. The program is modeled after the current NRC Examiners Standard (ES)-601 requalification examinations given at commercial reactor facilities.

The SRS peer evaluation program is outlined in "Savannah River Site Peer Evaluator Standards - Operator Assessment for Restart." The peer evaluation process, as implemented in November 1990, consisted of three parts over the period of 1 week. The first part was a 40-question, open reference, written examination developed and approved jointly by WSRC and DOE. The second part was an in-plant examination which consisted of 10 job tasks either performed or simulated in the K-Reactor or simulator. The operator was expected to complete the task using whatever tools would be available during normal operations. The last part of the evaluation was a day of simulator examinations. The examination consisted of three or more scenarios with various crew configurations. Two evaluations were performed as part of the simulator examination. The first was an individual evaluation based upon each person successfully completing the critical duties of his/her position. The second was a crew evaluation that judges the crew's ability to work as a team, communication between team members, and other soft skills such as procedure compliance and overall awareness. The overall process was structured such that WSRC conducts the evaluations and DOE independently evaluates the crews at the same time. The DOE evaluator team consisted of both DOE personnel and current NRC Licensing Examiners. The WSRC team was a group of hand-selected certified supervisors from P and L reactors who have been through an evaluator training program.

The subject material of the examination was derived from a Knowledge and Abilities (K&A) Catalog, revision 0, which was developed as part of the restart effort. The K&A Catalog defines knowledge and ability levels the operating crews are required to demonstrate at the time of restart. The

catalog is modeled after those developed for commercial reactors by the NRC. The levels subjected to testing in November 1990 were those items asterisked in Revision 0 of the K&A Catalog.

Qualification Cards

Another evaluation process established to determine operator and supervisor qualification levels is the qualification card program. The qualification card consists of three main sections: a study guide, a knowledge evaluation section, and a task demonstration section. The study guide section is a narrative description of the system, its purpose, design, and other relative material. The knowledge requirements section outlines those knowledge areas the individual is evaluated on and must successfully complete prior to completion of the qualification card. In many cases the qualification card knowledge requirements are expansions of the K&A catalog knowledge criteria. The demonstration requirements include such items as locating equipment, locating equipment controls and instrumentation, actual equipment manipulation, and performance of a procedure.

In addition to all the other reviews performed by DOE, the DOE staff reviewed selected WSRC closure packages associated with training issues and determined that all items had been adequately addressed and no additional issues remained unresolved.

10.2.4 Conclusion

The staff has reviewed the restart training program. With the exception of the open items identified, the DOE staff concludes that the restart training program meets the restart criteria. The resolution of these open items will be documented in a supplement to the SER.

10.2.5 References

1. WSRC, Reactor Operations Standing Instruction (ROSI) III.1, "Organization of Reactor Operations," Revision 5.
2. WSRC, DPSOP 38, "Reactor Personnel Selection, Qualification, and Training Manual," Revision 33.
3. WSRC, "Training Administrative Manual (TAM)," Revision 1.
4. WSRC, OPS-ROTP-890660, "Policy on Performance Evaluation and Testing Using the Simulator."
5. WSRC, ROSI V.7, "Onshift Training Program," Revision 0.
6. U. S. Department of Energy, DOE Order 5480.6, "Safety of DOE-Owned Reactors."
7. American Nuclear Society, Standard 3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants."

8. WSRC, DPSOL 100-4000 PLK, "Emergency Classification, Notification, and Initial Actions," Revision 4.
9. WSRC, RSES 109, "Required RSES Employee Training and Technical Staff Qualification," Revision 2.
10. Westinghouse, WISR 31, "Management Assessment Plan," dated December 15, 1989.

Attachment 10.2-1
K-Reactor Restart Training

SM/CRS/CCRO Restart Training

CYCLE1

Leak Before Break
Seismic
Operational Anomalies
Emergency Action Levels
Formality of Operations
Control of Combustibles
Shift Interface
Procedure Compliance
Logkeeping
Simulator Training

CYCLE2

Technical Specifications
Reactivity Assessment
Conduct of Operations
Industry Events
Core Physics
Simulator Training

CYCLE3

Formality of Work Practices
Team and Diagnostic Skills
Reactor Events
New Professionalism
Simulator Training

CYCLE4

Fire Protection
Emergency Action Levels
Reactivity Tracking
Linking Document
New Technical Specifications
Work Practices
Simulator Training

CYCLE5

On-The-Job Training
Moderator Recovery System
Lube Oil System
Kanne's
Ten Minute Line Monitor
Cooling Water Gamma Monitor

CYCLE6

New Technical Specifications
Non-Conformance Reporting
Teamwork and Diagnostic Skills
Event Reporting
Simulator Training
Assessment and Testing
108-Building
Valve Theory/Lubrication
Fundamentals

CYCLE7

Valve Fundamentals
Pump Fundamentals
Process Instrumentation
Simulator Training

CYCLE8

Technical Specifications
HP Enhancements
Reactivity Assessment
Simulator Training
Basic Electrical Theory
Diesels
108 Walkthrough

CYCLE9

Symptom-Based Procedures
Simulator Training
Math Fundamentals
Reactor Event Reports

CYCLE11

On-The-Job Training
Recirculation Pump Diesel Generator
Seismic Trigger
Reactor Tank Temp Meter
Seismic Lighting
Emergency Radios
Contaminated Water Holding Tank

CYCLE13

WISR-1, Bldg. 190 Pump Trips
WISR-37, CW/PW Leak Procedure Changes
WISR-2, ECS Flow Reduction

CYCLE15

Electrical Distribution (480 VAC and
125 VDC)
BTM's
APM's
RDSI for TPC's, Procedural Compliance
Simulator Training

CYCLE17

SCRAM Circuits
Electrical Review (Deleted)
Process Water System
Simulator Training

CYCLE10

Symptom-Based Procedures
Symptom-Based Procedure Simulator
Training
Math Fundamentals
Classical Physics

CYCLE12

Interpersonel Skills
WISR 7, 19, 35
P & ID's
REMACS
Simulator Training
Physics

CYCLE14

Startup Reactivity
New Technical Specifications
(6/89 Version)
T-Amps and New Procedures
Simulator Training

CYCLE16

Reactor Theory
Applied Reactor Theory
Simulator Training

CYCLE18

Peer Evaluation

CYCLE 19

Safety Computer
MC-29, VSF Status Trees
Conduct of Operations
Notification Procedure (RDP 12.02)
Introduction to New Technical
Specifications (REVD)
Simulator Training

CYCLE 21

Electrical Theory
Electrical Distribution (13.8 kV and
Mimic Bus)
Electrical Schematics
Simulator Training

CYCLE 23

Fire Protection
SSS Qual. Card
Control Rod Drive Qual. Card
Simulator Training

CYCLE 25

Startup Procedures
Simulator Training

CYCLE 20

Peer-Re-evaluations
Qualification Cards

CYCLE 22

EVAL (EPIP-6Q7-001, 002)
The 13 Shutdown Technical
Specifications (Rev. 0)
ECS Qual. Card
Fire Brigade
Simulator Training

CYCLE 24

Electrical ACC's (Station Blackout)
Simulator Training

CYCLE 26

Power Operations
Simulator Training

Attachment 10.2-2
K-Reactor Restart Training

Reactor Building Operator (RBO) Restart Training

CYCLE1

Leak Before Break
Seismic
Operational Anomalies
Emergency Action Levels
Formality of Operations
Control of Combustibles
Shift Interface
Procedure Compliance
Logkeeping

CYCLE3

Formality of Work Practices
Team and Diagnostic Skills
Reactor Events
New Professionalism

CYCLE5

On-The-Job Training
Lube Oil System
Kanne's
Cooling Water Gamma Monitor

CYCLE7

Valve Fundamentals
Pump Fundamentals
Process Instrumentation

CYCLE2

Technical Specifications
Formality of Operations
Industry Events
Core Physics

CYCLE4

Fire Protection
Emergency Action Levels
Reactivity Tracking
Operating Envelope (Linking Document)
New Technical Specifications
Work Practices

CYCLE6

New Technical Specifications
Teamwork and Diagnostic Skills
Event Reporting
Assessment and Testing
108-Building
Valve Theory/Lubrication Fundamentals

CYCLE8

Technical Specifications
HP Enhancements
Basic Electrical Theory
Diesels
CAT and GM Diesels
108 Walkthrough

CYCLE 9

Symptom-Based Procedures
Math Fundamentals
Reactor Event Reports

CYCLE 11

On-The-Job Training
Recirculation Pump Diesel Generator
New Equipment Installations

CYCLE 13

ECS Actuation Procedure
Physics
RO Manipulations/Blanket Gas System

CYCLE 15

RDSI's II.8, II.4
DPSOL 1067
Electrical Distribution
BTM's
Moderator Management
APM

CYCLE 17 (Qual. Card Title)

Blanket Gas Operation
Cell Flow
Coolant Return Tank
191 Building Booster Pump
Fuel Oil Distribution
Electrical Equipment
GM Diesel (Pre-2611)
Shield System
Reactor Room Floor Drain Pots
Sumps and Leak Collection

CYCLE 10

Math Fundamentals
Classical Physics

CYCLE 12

Interpersonel Skills
WISR 7, 19, 35
P & ID's
Physics

CYCLE 14

Procedure Changes
Purification
Purification OJT

CYCLE 16 (Qual. Card Title)

Distillation
Helium Distribution
Chilled Water
Disassembly Basin Filtration
Lube Oil Handling
Caterpillar Diesels
Compressed Air (Rev. 1)
Bingham Pump Seal System
Bingham Pump Motor
Bingham Pump Lube Oil

CYCLE 18 (Qual. Card Title)

BG Instrumentation
Conductivity Control
Disassembly Basin Management
Disassembly Basin Recirc.
Spray System
191 Building Standby Air
Ventilation and Fire Protection
Miscellaneous Equipment
CHR

CYCLE 19 (Qual. Card Title)

Process Water (A/D)
Fans and Filters (A/D)
105 Building Exhaust Fans and EG Sets
SSS

CYCLE 21

Electrical Science
Electrical Schematics
Fire Brigade

CYCLE 23 (Qual. Card Title)

TWFM
Cell Crane
Tritium

CYCLE 25 (OPEN)

CYCLE 20 (Qual. Card Title)

Actuator System (A)
Cooling Water (A/D)
ECS Addition/Mix Polybor
Deion. Makeup/Deut. (K)
Disassembly Basin Deionizers

CYCLE 22

Ham (K)
SSS Ink Prep. (K)
Stack Samples/STM's
Inhibitor System (K)

CYCLE 24 (Qual. Card Title)

Essential Equipment Checks
Moderator Handling
Drain/Refill PW System

CYCLE 26 (OPEN)

Attachment 10.2-3
K-Reactor Restart Training

Shift Technical Engineer (STE) Program

PART 1

- STE Roles and Responsibilities
- STE Roles and Lessons Learned
- Thermal and Hydraulic Sciences
- Reactor Limits and Safety Analysis
- Applied Reactor Physics
- Safety Envelope and Bases
- Symptom-Based Procedures Overview
- Charge and Discharge Operations
- Reactor Startup

PART 2

- Accident and Transient Analysis
- Mitigating Core Damage
- Additional Applied Reactor Physics
- **TEST OUT** of the following **SRQ** Fundamentals:
 - Nuclear Science
 - Material Science
 - Radiation Protection
 - Chemistry
 - Print Reading
 - Electrical Science
 - Thermodynamics and Fluid Flow

PART 3

- DPSOL 105-1060A, ECP and Rod Moves
- STE Criticality Safety
- INPO Observation Techniques
- NCR's
- USQD's
- Practical Problems - calculating effects on plant parameters after various scenarios occurred

PART 4

- Thermodynamic Science
 - Fluid Flow
 - Reactor Heat Transfer
 - Thermo-Hydraulic Limits
- Mechanical Science
- Instrumentation and Controls
- Chemistry
- Radiation Protection
- Material Science
- Nuclear Science
- STE Applied Fundamentals
 - Classroom
 - Simulator

10.3 Normal and Abnormal Operating Procedures

10.3.1 Background

Procedures are a key factor affecting operator performance. The probability of operator error increases greatly with the use of poorly written procedures. In addition, deficient procedures and failure to follow procedures are major contributors to the occurrence of many significant operational events. Properly controlled and readily available procedures promote their use and ensure operational activities will be conducted in the manner intended.

The Savannah River Site (SRS) production reactors have been operated by procedures that were not properly maintained and which were not updated to meet commercial nuclear industry standards. Inadequate procedure content and administrative control have been found to be contributors of several events at the SRS. WSRC committed to issue new or revised operating procedures that address operator actions to limit severe accidents.

10.3.2 Restart Criteria

The criteria for determining the adequacy of the SRS operating procedures for restart are described below:

- (1) Activities at SRS reactors shall be accomplished for reactor restart, and for the remainder of SR reactor plant life, in accordance with approved plant procedures. Procedures shall be prepared, reviewed, and approved prior to implementation. These procedures shall include appropriate quantitative and qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished and shall provide an approved preplanned method of conducting operations. These acceptance criteria shall be met for each reactor before restart of that reactor.
- (2) For reactor restart, procedures shall be established in the following areas:
 - o Administrative Procedures
 - o Abnormal, Off-Normal, or Alarm Condition Procedures
 - o Maintenance Procedures - see Section 13, "Maintenance"
 - o Radiation Control Procedures
 - o Chemical and Radiochemical Control Procedures
 - o Emergency Operating Procedures - see Section 10.4, "Emergency Operating Guidelines and Procedures"
 - o Emergency Plan Implementing Procedures - see Section 15, "Emergency Preparedness"

- o Modification Procedures
 - o Reactivity Manual
 - o Calibration, Surveillance Test, Test and Inspection Procedures (including control of measuring and test equipment)
- (3) Attachment 1 to this SER section contains the procedural action items (most of which are identified in the ROMP) that are required to be established prior to reactor restart. Procedures or procedural revisions shall be established to implement these action items.
- (4) A procedures program that includes the following shall be implemented:
- o A verification/validation program that includes the following:
 - A review effort that ensures procedures are technically adequate and workable.
 - A procedure walkdown process by qualified personnel.
 - An independent review group that performs a sample review of revised procedures.
 - o A qualified reviewer program that provides assurance that procedures or procedure changes that may constitute an unreviewed safety question are appropriately identified to management. This program shall include qualification training for reviewers. Prior to restart, this program shall be established and training shall have been initiated and scheduled for post restart completion.
 - o A procedure review and approval process that includes technical review, cross-disciplinary reviews, and management approval. Documentation of this process shall be maintained.
 - o A non-intent "temporary-change-to-procedure" program that ensures the following:
 - The intent of the original procedure is not altered.
 - The change is approved by two members of the plant management staff, at least one of whom is a certified supervisor.
 - The change is documented and approved by a designated management position within 14 days of issuance of the change.

- o A review and revision program for approved procedures shall be initiated at restart. Scheduling of procedure reviews/revisions shall be prioritized such that the important safety-related procedures are reviewed every 2 years, and every procedure is reviewed every 5 years.
- (5) The procedures listed in Attachment 1 to this section of the SER shall be processed through the new procedure program. In addition, when existing procedures must be revised, revisions shall be made using the new procedural program process.
 - (6) Reactor Operations Department, Power Department, and Central Services Works Engineering (CSWE) safety-related procedures (See SER section 5.6, Table 5.6-1) that are not addressed in Attachment 1, but will be used in the field when a reactor is restarted, shall be evaluated to ensure that these procedures are technically adequate and workable prior to use. Sampling techniques may be used to obtain the level of confidence necessary for restart.
 - (7) For reactor restart, programs shall be established for the following areas:
 - o General plant procedures for startup, shutdown, power operations, process monitoring, and fuel/target handling
 - o System operating procedures
 - (8) For reactor restart, a program shall be established to ensure that reactor operational commitments that have been addressed by specific procedures are tracked so that future revisions and upgrades are not negated in the process.

10.3.3 Evaluation

The restart criteria fall into two distinct groups, each with a specific purpose. The first group of criteria was designed to assure a structured process was established for writing, revising, reviewing, and approving procedures. All of the 10.3 SER restart criteria except criterion 3 and criterion 5 are directed toward this goal.

Restart Criteria 3 and 5 require procedure development for, or revision to, the programs listed in Attachment 1. The items listed in Attachment 1 are specific Charlotte and WISR commitments, which are identified to assure that these activities are tracked and included in the procedures work scope.

General policy guidance on the need for, and use of, procedures is contained in several documents. ROSI II.8, "Procedure Compliance" (Reference 1), requires that all activities shall be performed in accordance with approved procedures. Likewise, Westinghouse Savannah River Company Quality Assurance Manual 1Q, Procedure QAP 5-1, "Instructions, Procedures, and Drawings" (Reference 2), specifies that activities affecting quality shall be prescribed by and performed in accordance with documented procedures.

Management Requirements and Procedures (MRP) Manual 1B, Procedure MRP 3.27, "Management of Division and Department Level Procedures" (Reference 3), and DMP 7.01, "Development and Control of Reactor Operations, Components Handling, and Maintenance Procedures" (Reference 4), specify the requirements that must be met in preparing, revising, and approving procedures (including the sequence that must be followed). The following is a summary of the process followed in preparing or revising a procedure. The writer prepares the initial draft of the new procedure or procedure revision. The draft is then sent out for technical and multidiscipline reviews, as required. Any comments are then incorporated by the writer. The draft is then walked down by a qualified walkdown validator. His comments are incorporated by the writer. Following resolution and incorporation of all comments, the draft goes to the Operations and Maintenance Procedure Review and Approval Committee (OMPRAC), a subcommittee of the Reactor Operations Safety Committee (ROSC), for review and approval. If not approved, the draft goes back to the writer to resolve comments or rewrite the procedure, as required. If approved, any required training is performed on this procedure or procedure change. Following completion of training, the procedure is then implemented.

DPSP-85-1090, "Procedure Writer's Guide" (Reference 5), provides guidelines for (1) preparing and revising reactor operations and maintenance procedures; (2) planning, drafting, and self-reviewing procedures and their revisions; (3) conducting procedure validations; and (4) performing field evaluations of procedures.

As discussed above, new procedures or procedure changes receive technical reviews, cross-discipline reviews (if required by DMP 7.01), and walkdown validations. The technical reviews are required to be performed by personnel qualified as Qualified Reviewers. In addition to performing an in-depth technical review of the procedure, RDP 12.03, "Unreviewed Safety Question Initial Screening and Evaluation Procedure" (Reference 6), requires the Qualified Reviewer to verify the USQ determination that the procedure originator completes. DMP 7.06, "Technical Procedure Reviewer Qualification" (Reference 7), requires all technical reviewers to possess a 4-year degree in engineering or a related science and have a minimum of 3 years nuclear experience or be a certified senior reactor operator or have a total of 8 years related nuclear experience. In addition, technical reviewers must complete qualified reviewer training and successfully pass an interview by the Procedures Manager or his designee. Qualified reviewers must recertify every 2 years. Similar to the requirement for reviewers to be qualified, personnel designated as walkdown validators must also possess a minimum level of education and training. ROSI V.6, "Walkdown Validator Qualification Program" (Reference 8), defines the requirements for qualification as a validator and ensures that personnel performing walkdowns are capable of performing their duties. Validators must possess at least a high school diploma and either be a certified operator at SRS or have a minimum of 2 years experience as a commercial operator or Navy nuclear plant operator. These validators shall be trained on the ROSI V.6 procedure validation checklist, shall receive a plant formalization walkthrough, and shall read required material from the facility study guide. These individuals must recertify every 2 years.

In addition to the technical reviews and walkdowns, the procedures receive an additional review by OMPRAC, which is the multidiscipline review board for final procedure reviews. OMPRAC is the approval authority for operating and maintenance procedures. OMPRAC is not a part of any plant line organization; therefore, it performs review and approval functions that are independent from those performed by the line organizations during their reviews of the procedures. DMP 7.02, "Operations and Maintenance Procedure Review and Approval Committee" (Reference 9), directs the board members to review the procedures for technical accuracy, human factors, writing style, user and plant compatibility, temporary procedure change (TPC) criteria, and equipment and compliance criteria. In addition, DMP 7.02 directs the board to ensure prior to approval that the procedures do not contain USQs and that, if questions do exist, the procedure is to be forwarded to the Reactor Independent Safety Review Committee (RISRC) for resolution.

Temporary procedure changes (TPCs) are controlled by ROSI II.17, "Temporary Procedure Changes" (Reference 10). TPCs are authorized procedure changes initiated when unforeseen situations require prompt action to ensure reactor safety or to correct an operating deficiency. ROSI II.17 describes the review and approval process for TPCs. This process basically consists of two steps. The first being initial temporary approval, and the second being final approval. TPCs may be implemented following the temporary approval of the TPC. Final approval of the change must then occur within 7 calendar days of the TPC implementation date, although the Technical Specifications allow 14 days for this final approval. The ROSI specifies the required approval authorities for both non-intent TPCs and intent changes. The non-intent TPC program meets the restart criteria, which require that (1) the intent of the original procedure not be altered; (2) the change be approved by two members of the plant management staff, at least one of whom is a certified supervisor; and (3) the change is documented and approved in accordance with Technical Specifications. If the change involves a change of intent to the procedure, a full review and approval in accordance with Technical Specifications must occur before the change may be implemented. All TPCs are to receive an initial USQ screening prior to approval. TPCs that constitute a USQ are not to be processed.

DMP 7.04, "Periodic Review of Reactor Operations, Components Handling, and Maintenance Procedures" (Reference 11), addresses the requirements for periodic reviews of procedures. The restart criteria require that safety-related procedures be reviewed every 2 years and that all other procedures be reviewed at least every 5 years. These requirements are met by DMP 7.04.

Safety-related procedures that are used in the field when a reactor is restarted have been walked down to determine if they are technically adequate and workable. RTAP has developed a list of safety-related procedures that are required to be verified for restart. The procedures are in the process of being revised as determined by the walkdown process. Any comments that are generated as a result of the walkdowns are dispositioned in one of two ways. If the comment is technical in nature or affects the workability of the procedure, the comment must be incorporated into the procedure prior to restart. If the comment is non-technical and does not affect the workability

of the procedure (e.g. minor format change), the comment will be tracked until it is incorporated, but is not required to be incorporated prior to restart. The revisions are to be handled in accordance with the criteria of DMP 7.01. The program to review, revise, and approve the safety-related procedures meets the requirements of the restart criteria.

DMP 7.01 specifies how all reactor operations, components handling, and maintenance procedures are to be controlled. This process includes establishing a tracking program that will monitor the status of a procedure from initiation of development to final approval and distribution of the procedure, specifying the review and approval requirements (as discussed earlier in this evaluation section), establishing training requirements for the procedure, tracking periodic review dates for the procedure, and ensuring that all appropriate documentation is maintained in accordance with appropriate QA requirements. All procedures that are revised at their normal review and revision cycle, and those revised to incorporate requested changes are to be handled in accordance with this program.

Power Department procedures are handled in accordance with PDAP 5-3, "Operating Procedures" (Reference 12). This procedure establishes the responsibilities and defines requirements for the development, review, approval, and use of operating procedures for the Power Department. The procedure defines the need and requirements for qualified reviewers, approvals, and temporary changes. PDAP 5-3.1, "Independent Qualified Reviewer" (draft) (Reference 13), establishes the requirements for independent reviewers and PDAP 5-3.2, "Verification/Validation of Operating Procedures" (draft) (Reference 14), establishes the requirements for verification and validation of Power Operations Procedures. The Power Department Procedure Writer's Guide has been developed and is currently in draft form. When approved, these procedures will comprise the Power Department procedure program and will meet the restart criteria. Pending completion and approval of these procedures, this issue will remain an open item.

Central Services Works Engineering (CSWE) procedures are handled in accordance with DPSPM-GEN-52, "CSWE Administrative Procedures" (Reference 15). This procedure comprises the program for procedures for CSWE. The procedure includes the requirements for preparation, technical review, walk throughs, validation, revisions, temporary changes, approvals, and periodic reviews. The program has been evaluated and satisfies all the restart criteria.

One criterion was added to the restart criteria after several administrative procedures were found to have been modified after closure of the SER program reviews. Although modifications to administrative procedures are not unusual, the fact that several changes altered or deleted sections that implemented restart criteria highlighted the immediate need for a program that assures previous commitments are not inadvertently modified or deleted. WSRC does not currently have a formal program for tracking and reviewing commitments in procedures as a part of the revision, review, and approval process, although work has begun in this area. Pending the development of such a program, commitment tracking remains an open item.

The staff determined that the WSRC procedure program does not fully meet the restart criteria in that the administrative procedures do not specify the categories of procedures that are required for restart. Procedures actually exist in all areas except for general plant procedures and system operating procedures. Pending the establishment of administrative controls that define the categories of procedures and ensure that procedures are developed in all required areas, this issue remains an open item.

WSRC is in the process of developing General Operating Procedures (GOPs) and System Operating Procedures (SOPs). GOPs will provide structured steps for overall plant operation and direct the performance of other procedures. SOPs provide detailed steps for operating plant systems under normal operation including placing systems in service and removing systems from service. A procedure writers guide is currently being drafted for these new types of procedures. This guide will reflect existing commercial standards based on NRC and INPO documents. Until such time as GOPs and SOPs are implemented, the plant will be operated using the existing DPSOL system.

Attachment 1 to this section contains the procedural action items that are to be established prior to reactor restart. Attachment 2 lists the action items from Attachment 1 and the program, procedure, or process WSRC used to address each item. The adequacy of the individual procedures and programs has not yet been reviewed by the DOE staff. The staff will report on the results of that review in a supplement to this SER. This is an open item.

10.3.4 Conclusion

The staff has identified a number of open items in the procedures upgrade program as identified above. Upon satisfactory resolution of those deficiencies, the staff will be able to conclude that the procedures upgrade program is sufficient for restart. The resolution of those open items will be addressed in a supplement to the SER.

10.3.5 References

1. WSRC, Reactor Operations Standing Instructions, Procedure II.8, "Procedure Compliance," Rev. 7.
2. WSRC, Quality Assurance Manual 1Q, QAP 5-1, "Instructions, Procedures, and Drawings," Rev. 0.
3. WSRC, Management Requirements and Procedures Manual 1B, MRP 3.27, "Management of Division and Department Level Procedures," Rev. 0.
4. WSRC, Manual RD 7.1, Procedure DMP 7.01, "Development and Control of Reactor Operations, Components Handling, and Maintenance Procedures," Rev. 0.
5. WSRC, Manual RD 7.1, Procedure DPSP-85-1090, "Procedure Writer's Guide," Rev. 9.

6. WSRC, Manual RD-1, Procedure RDP 12.03, "Unreviewed Safety Question Initial Screening and Evaluation Procedure," Rev. 0.
7. WSRC, Manual RD 7.1, Procedure DMP 7.06, "Technical Procedure Reviewer Qualification," Rev. 1.
8. WSRC, Reactor Operations Standing Instructions, Procedure V.6, "Walkdown Validator Qualification Program," Rev. 0.
9. WSRC, Manual RD 7.1, Procedure DMP 7.02, "Operations and Maintenance Procedure Review and Approval Committee," Rev. 1.
10. WSRC, Reactor Operations Standing Instructions, Procedure II.17, "Temporary Procedure Changes," Rev. 0.
11. WSRC, Manual RD 7.1, Procedure DMP 7.04, "Periodic Review of Reactor Operations, Components Handling, and Maintenance Procedures," Rev. 0.
12. WSRC, Power Department Manual, PDAP 5-3, "Operating Procedures." Rev. 2.
13. WSRC, Power Operations Administrative Procedure PDAP 5-3.1, "Independent Qualified Review" (draft).
14. WSRC, Power Operations Administrative Procedure, PDAP 5-3.2, "Verification/Validation of Operating Procedures" (draft).
15. WSRC, Manual DPSPM-GEN-52, "CSWE Administrative Procedures," Rev. 2.

ATTACHMENT 1

Minimum Reactor Operations Management Plan (ROMP) affected procedures or procedural revisions that are required to be established prior to restart as follows:

<u>ITEM #</u>	<u>ROMP AREA</u>	<u>SUBJECT</u>
Administrative Procedures		
1.		Control Room Access
2.	MS-7	Shift Turnover
3.	MS-10	Housekeeping
4.	MS-5.3	Operating procedures TPC reduction
5.	DB-1.4	Technical Specifications on core power
6.	DB-2.4	Technical Specifications on leak limits
7.	QA-5.1	Instructions Procedures and Drawings
8.	QA-6.1	Document Control
9.	MS-9.1	Control of Testing
10.	MS-12	Organization Responsibilities
11.	MS-19	Unreviewed Safety Questions
12.	MS-21	Design Control
13.	MS-22	Design Bases Test
14.	WISRC-23	Core Evaluation Response
15.	WISRC-33	Plant Site Organization
16.	MS-29	Operating Experience Review
17.	MS-20	Communications (technical)
18.	MS-4	Communications
19.	MS-17	Post Trip assessment and review
20.	MS-16	Records Management
21.	MS-2	Tagouts and Lockouts
22.	DB-4	Fire Protection
23.	MS-1	Logkeeping
24.	WISRC-3	ECS Performance Testing
25.	WISRC-4	LOCA Performance Testing
26.	WISRC-21	Spurious Controls Operation
27.	DB-5	Power Limits
28.	WISRC-31	Operational Assessment
29.	WISRC-34	Confinement Performance
30.	WISRC-40	Root Cause Analysis
31.	MS-9.2	Test Procedures
32.	MS-14	Compliance
33.	WISRC-39	Procedure Compliance
34.	MS-5.2	Procedural Revision Event report
35.	MS-5.4	Procedure walkdown reviews
36.	MS-15.1	Categorize Backlog items
37.	QA-5.2	Instructions Procedures and Drawings
38.	STP	Startup Test Program

<u>ITEM #</u>	<u>ROMP AREA</u>	<u>SUBJECT</u>
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General Plant Procedures for startup, shutdown, power operations, process monitoring, and fuel/target handling

39.	DB-2.11	Pre-startup Seismic walkdown
40.	DB-3.3	Reactivity Control Procedures
41.	MS-5.1	Procedure Review
42.	MS-6.1	Significant Safety Equipment List
43.	MS-8.1	Up-to-date DPSOL schematics
44.	MS-8.2	Independent verification
45.	DB-3.4	Surveillance procedures

System operating procedures. instructions for energizing, filling, venting, draining, starting up, shutting down, changing modes of operation, returning to service following testing (if not contained in test procedure), and other instructions appropriate for operation of systems shall be delineated in system procedures.

Abnormal, Off-Normal, or Alarm Conditions Procedures

46.	DB-4.10	Shutdown on confirmed fire
47.	HP-7.2	Training topics
48.	MS-5.3	Review of TPCs

Maintenance Procedures

49.	MS-6.2	Safety Related Maintenance
50.	WISRC-8	Valve Inspection
51.	WISRC-18	Batteries and UPS

Radiation Control Procedures

52.	MS-11	Radiation Protection Practice
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Chemical and Radiochemical Procedures

53.		Moderator and Disassembly sampling
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Emergency/Significant Event Procedures

54.	DB-4.8	Fire protection/suppression
55.	WISRC-1	ECS actuation (LOPA)
56.	WISRC-2	ECS actuation
57.	WISRC-5	Moderator Recovery System
58.	WISRC-19	Essential Core Monitoring
59.	WISRC-37	Improved training Procedures
60.	WISRC-38	LOPA Event Recovery

<u>ITEM #</u>	<u>ROMP AREA</u>	<u>SUBJECT</u>
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Modification Procedures

Procedures that are written to address modifications:

61.	DB-2.2	Seismic Resistance
62.	DB-2.6	Seismic Qualified Makeup
63.	DB-2.8	Seismic Qualified-Insts
64.	DB-2.10	Seismic lighting
65.	OW	Outage work
66.	OG	Ongoing Projects
67.	WISRC-7	Sump water removal
68.	WISRC-21	Spurious Controls

Modification Procedures (cont.)

69.	WISRC-35	105 Building Exhaust fan design
70.	WISRC-49	Assimilation of massive changes

Reactivity Manual

71.	Reactivity Manual	
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Calibration, Test, and Inspection procedures including control of measuring and test equipment.

72.	QA-11.1	Test control
73.	WISRC-4	LOCA performance testing
74.	WISRC-6	Pump degradation

Calibration, Test, and Inspection procedures including control of measuring and test equipment (cont.)

75.	WISRC-18	Batteries UPS
76.	WISRC-38	LOPA Event frequency
77.	WISRC-39	Procedural Guidance

ATTACHMENT 2

This attachment specifies the closure documentation associated with each of the items identified in Attachment 1.

ITEM #	CLOSURE DOCUMENTATION
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1.	Control Room Access requirements are specified in Reactor Operations Standing Instruction (ROSI) III.15, "CCR/KCR-Nonassigned Personnel Control" and the Conduct of Operations Manual, R2.1. For additional information on these requirements, refer to SER section 10.15.
2.	Shift Turnover requirements are specified in ROSI III.3, "Shift/Assignment Turnover." For additional information on this subject, refer to SER 10.7.
3.	The Housekeeping requirements are specified in RDP-3.02, "Housekeeping/Material Condition Control." For further information on this subject, refer to SER section 10.5.
4.	Temporary procedure changes are handled in accordance with ROSI II.17, "Temporary Procedure Changes." For further information on this subject, refer to the evaluation section of this SER section.
5.	Technical Specifications on core power - 41 procedures were identified for revision in response to this item. These revisions have been completed.
6.	Technical Specifications on leak limits - 12 procedures were identified for revision in response to this item. These revisions have been completed.
7.	Instructions, Procedures and Drawings - DMP-7.01, "Development and Control of Reactor Operations, Components Handling, and Maintenance Procedures" was revised to address this issue.
8.	Document Control - ROSI II.6, "Receipt and Processing of Procedure Revisions, Changes and New Procedures," ROSI II.8, "Procedure Compliance," and a Dupont Standard Operating Log (DPSOL) were revised in response to this issue.
9.	Control of Testing - Category 5 Technical Specifications addresses this issue of approval of test authorizations which fall outside of Tech Specs. The implementing procedure for this requirement is ROSI II.12.

- ITEM # CLOSURE DOCUMENTATION
10. Organizational Responsibilities - Category 5 Tech Specs outline the organizational responsibilities for Westinghouse Savannah River Company (WSRC). Various department level implementing procedures specify the organizations and responsibilities at the department level (for example the reactor Operations Department organization and responsibilities are outlined in ROSI III.1).
 11. Unreviewed Safety Questions - Category 5 Tech Specs include requirements on defining Unreviewed Safety Questions (USQs), defining responsibilities associated with the review of activities for potential USQs, and specifying the process for disposition of USQs. The implementing procedure, RDP 12.03, "Unreviewed Safety Question Initial Screening and Evaluation Procedure," describes this process in detail.
 12. Design Control - This issue is discussed in Section SE-5.4.
 13. Design Bases Test - This issue was closed upon issuance of Category 5 Tech Specs.
 14. Core Evaluation Response - This issue was covered by issuance of Category 5 Tech Specs and the revision of 2 DPSOLs.
 15. Plant Site Organization - ROSI III.1, "Organization of Reactor Operations" and a management directive were issued to address this issue.
 16. Operating Experience Review - RDP 9.06, "Operating Experience Review" is the administrative procedure which describes the overall process of operating experience review and feedback. For further information on this subject, refer to SER section 10.16.
 17. Communications (technical) - Several documents were issued to address this concern. They included Category 5 Tech Specs, Management Requirements and Procedures (MRPs) 4.07 through 4.10, and Savannah River Laboratory procedure DPST-89-368, "Immediate DOE Notification Procedure."
 18. Communications - ROSI IV.8, "Communications" implements the communications guidelines and practices outlined in the Institute of Nuclear Power Operations (INPO) guidelines. RDP 12.02, "Notification of Reactor Plant Conditions, Events, and Concerns" contains the procedural guidance concerning reporting requirements.

ITEM # CLOSURE DOCUMENTATION

19. Post Trip assessment and review - ROSI II.16, "Reactor Scram/Unscheduled Shutdown Recovery" specifies the requirements for documenting the investigation and determination of the cause, documenting restart approvals, and obtaining the proper concurrences for restart.
20. Records Management - Reactor Technology Report (RTR) 2627 and 2 DPSOLs were issued/revised to address this issue.
21. Tagouts and Lockouts - RD 3.1, DMP 3.1, Appendix A, "Lockout" contains the requirements associated with tagging and locking components. For additional information on this subject, refer to SER section 10.9.
22. Fire Protection - 40 DPSOLs were developed to resolve this concern.
23. Logkeeping - ROSI II.1, "Shift Operating Logs" and the Conduct of Operations Manual, R2.1 together specify the requirements associated with logkeeping. For further information on this subject, refer to SER section 10.6.
24. ECS Performance Testing - This item has been evaluated and determined to be adequately addressed by section DB-5.5 of the ROMP. Satisfactory closure of that section of the ROMP will satisfy the requirements of this section.
25. LOCA Performance Testing - Special Procedure (SP)-2447, "L-Reactor Special Hydraulic LOCA Tests" and two test authorizations TA 1-2293 and 1-2293A were generated to resolve this issue.
26. Spurious Controls Operation - This issue was evaluated and documented in Reactor Technology Memorandum (RTM)-4972. No procedures were generated as a result of this evaluation.
27. Power Limits - The results of this program will be documented in The Core Operating Report.
28. Operational Assessment - The Management Assessment Plan contains the requirements for the performance of senior management evaluations of the capabilities of the operating crews. For further information on this subject, refer to SER section 10.2.
29. Confinement Performance - Based upon technical review, no procedure revisions are required for WRSC 34.
30. Root Cause Analysis - The implementing procedures for this item are RDP 9.02, "Occurrence Reporting System," dated January 1, 1991, and WSRC-IM-91-3, "Root Cause Analysis Handbook."

ITEM # CLOSURE DOCUMENTATION

31. Test Procedures - The requirements specified in this section are contained in ROSI II.12, "Test Control."
32. Compliance - ROSI II.8, "Procedure Compliance" and DPSOL 338-205-1, "Instructions, Procedures, and Drawings" specify procedure-in-hand requirements and other aspects of procedure compliance.
33. Procedure Compliance - 2 DPSOLs were revised in order to close this issue.
34. Procedural Revision Event Report - 27 DPSOLs were revised in order to close this issue.
35. Procedure Walkdown Reviews - As a result of the walkdowns performed, 103 DPSOLs were identified as requiring revision in order to close this issue. This work has been completed.
36. Categorize Backlog Items - The reviews done in accordance with this ROMP item identified approximately 200 DPSOLs which needed to be developed or revised. These procedures/procedure revisions are still in progress.
37. Instructions Procedures and Drawings - Two ROSI's were revised in accordance with the requirements of this item; ROSI II.6 and ROSI II.8.
38. Startup Test Program - This item is covered in the Startup Test Program Master Plan, OPS-STP-89001.
39. Pre-startup Seismic Walkdown - 2 DPSOLs were issued to provide instructions for performing the walkdowns required by this section.
40. Reactivity Control Procedures - 30 DPSOLs were issued in response to this issue. These procedures were necessary to enhance reactivity monitoring and ensure proper response to reactivity anomalies.
41. Procedure Review - In response to the reactivity event at P-Reactor, 30 DPSOLs were revised or created to close this issue.
42. Significant Safety Equipment List - DPSOL 105-1829A was revised to include the identified safety related systems. Additional requirements for safety-related equipment lists are identified in Section 5.6 of this report.
43. Up-to-date DPSOL Schematics - 197 DPSOLs were revised to update the DPSOL schematics in accordance with the guidance of this commitment.

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- 44. Independent Verification - ROSI IV.15, "Independent Verification" and RD 3.1, DMP 3.1, Appendix A, "Lockout" specify the independent verification requirements for all of the occasions where independent verification is required. In addition, there are a number of associated procedures that contain various requirements associated with the process of independent verification. For further information on this subject, refer to SER section 10.12.
- 45. Surveillance Procedures - 4 DPSOLs were revised/developed to address this issue.
- 46. Shutdown and Confirmed fire - 11 DPSOLs were developed/revised to address responses to fires in the reactor facility and upon confirmation of a fire, to place the reactor in a shutdown condition.
- 47. Training Topics - In response to the training requirements identified in this section, 113 DPSOLs were identified as requiring either revision or development. Additional information on training is addressed in SER section 10.2.
- 48. Review of TPCs - Item 4 above discussed the revision of ROSI II.17 to control the process of generating TPCs. In addition to this, procedure reviews were performed to determine the number of existing procedures that needed revising to meet the criteria specified in this section. 54 DPSOLs were revised/developed as a result of these reviews.
- 49. Safety Related Maintenance - As a result of reviews in this area, approximately 350 procedures were identified as requiring revision. This work is ongoing.
- 50. Valve Inspection - RTM 4987 defined the valves that needed to be inspected. 5 DPSOLs were revised in the process of performing these inspections.
- 51. Batteries and UPS - 46 DPSOLs were identified as needing development or revision to satisfy this issue. That work is ongoing.
- 52. Radiation Protection Practice - DPSP 87-1211, section RPSP 102, "Radiation Protection-Transition Points" was the only document generated in response to this issue.
- 53. Moderator and Disassembly Sampling - 13 DPSOLs were identified as needing revision in response to this concern.
- 54. Fire Protection/Suppression - 3 DPSOLs were revised in order to meet the objectives of this item.

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55. ECS Actuation (LOPA) - 54 DPSOLs were revised/developed as a result of the studies and modifications associated with the 190 pump trip scenario.
56. ECS Actuation - ROSI II.17, "Temporary Procedure Changes" specifies that permanent TPCs for ACC procedures shall be incorporated into a procedure change, procedure revision or cancelled within 30 days from the effective date of the TPC. For further information on this subject, refer to the evaluation section of this SER section.
57. Moderator Recovery System - 5 DPSOLs were developed/revised as a result of the analysis on PW system leakage following a seismic event.
58. Essential Core Monitoring - Reactor Technology Report (RTR)-2662, "Minimum Essential Shutdown Core Monitoring" addresses this issue.
59. Improved Training Procedures - 29 DPSOLs were developed/revised to mitigate the consequences of severe accidents.
60. LOPA Event Recovery - 2 DPSOLs were revised/developed to incorporate new inspections identified in a study of current Cooling Water System inspections.
61. Seismic Resistance - 10 DPSOLs were generated to address performance testing of SSS seismic triggers.
62. Seismic Qualified Makeup - 16 DPSOLs were revised/developed to support performance testing of the Moderator Recovery system.
63. Seismic Qualified Instruments - 14 DPSOLs were revised/developed in support of performance testing of instruments.
64. Seismic Lighting - 2 DPSOLs were revised/developed in support of performance testing of lighting.
65. Outage Work - 24 DPSOLs were revised/developed in response to the concerns identified in this section of the ROMP.
66. Ongoing Projects - Quality Assurance Procedure (QAP) 9-1, "Control of Processes" and ROSI IV.11, "Control of Work" specify the requirements necessary to close this issue.
67. Sump Water Removal - 8 DPSOLs were revised to reflect changes made following plant modifications associated with sump water removal capability following LOCAs.

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68.	Spurious Controls - Reactor Technology Memorandum (RTM) 4972 contained the analytical assessment dealing with this topic. No procedures were identified as deliverables for this item.
69.	EP 903 Fan Design - As a result of design reviews and subsequent modifications, 5 DPSOLs were revised.
70.	Assimilation of Massive Changes - The closure of this item is documented in several documents; OPS-OAM-90-582, "Assimilation of Massive Change-Restart Item Work Plan, OPS-SAM-890008, "ORR Plan," OPS-SAM-890009, "ORR Procedure," The Management Assessment Plan, and various memos.
71.	Reactivity Manual - This issue is addressed in RTR 2595.
72.	Test Control - Based on review of functional tests, 18 DPSOLs were identified as requiring revision and have been revised.
73.	LOCA Performance Testing - See item 25.
74.	Pump Degradation - Based on studies of ECS, CWS, and sump pumps current surveillances on flow degradation, 3 DPSOLs were revised.
75.	Batteries, UPS - See item 51.
76.	LOPA Event Frequency - See item 60.
77.	Procedural Guidelines - See item 33.

10.4 Emergency Operating Guidelines and Procedures

10.4.1 Background

During high-stress conditions an operator needs quick, simple, concise directions. Emergency Operating Procedures (EOPs) provide operators with the appropriate actions/directions to prevent and mitigate the consequences of transients and accidents that cause plant parameters to exceed reactor protection system setpoints, engineered safety feature setpoints, or other appropriate technical limits.

One of the factors recognized by both DOE and WSRC as requiring upgrading prior to restart was the condition of the existing EOPs. At the time of the reactor shutdown, the operating contractor had been developing symptom-based EOPs for implementation in the field at a later date. All three production reactors were still operating using the existing Abnormal Condition Control Procedures and the 4000 Series DPSOLs (emergency procedures), neither of which were symptom-based or in the flowpath format used more recently in the commercial nuclear industry.

WSRC recognized the safety significance of EOPs and committed to developing a set of critical safety function status trees to link instrumentation with symptom-based procedures for the most critical safety functions, including subcriticality, core cooling, heat sink, confinement, and PWS inventory. WSRC also committed to revising the Emergency Cooling System (ECS) procedures based on the report on verification of acceptability and prioritization of ECS throttling limits.

10.4.2 Restart Criteria

The criteria for determining the adequacy of the SRS emergency operating procedures for restart are described below.

- (1) The methodology used to convert material provided by Emergency Operating Guidelines (OPS-ROTD-89-0486, "Development and Implementation of Symptom-Based Procedures for the SRS", (Reference 1)) into Emergency Operating Procedures (EOPs) shall be clearly defined.
 - (a) Engineering data that are used in the EOPs that were not introduced through the Abnormal Condition Control (ACC) procedures technical bases must be clearly defined.
 - (b) Ensure that all numerical values in the EOPs are adequately justified either in the ACCs Technical Bases or other documentation.
- (2) A writer's guide or equivalent controlled document for developing EOPs shall be approved and implemented.

- (3) An EOP Verification Program that confirms the correctness of EOPs and ensures that applicable technical information has been incorporated properly with human factors issues being considered shall be approved and implemented.
- (4) An EOP Validation Program that determines that the actions specified in the procedure can be performed by the operator to manage or mitigate the emergency condition effectively shall be approved and implemented. Action to be performed within 10 minutes of initiation of the event must be verified by simulator drills. System performance attributes of EOPs not validated by the simulator shall be identified, evaluated, and addressed.
- (5) Prescribed written guidelines that provide administrative controls over the following activities shall be in place:
 - o EOP preparation, review and approval, distribution, and revision
 - o EOP supporting documentation control
 - o EOP experience feedback
- (6) The EOPs are not required to be issued and implemented prior to restart. However, in the interim, WSRC will demonstrate that the ACCs are adequate for response to accident conditions.

10.4.3 Evaluation

Reactor Technology Paper DPST 89-291, "Vital Safety Function Status Trees" (Reference 2), specifies the methodology to be used in developing the EOPs. The paper sets the Status Tree priorities based directly on the fission product barrier concept, similar to the concept used in commercial nuclear power plants. The concept places the severity of the challenge in order of priority. The priorities are color-coded to provide easier recognition by the operator and, therefore, increase the response to the condition.

RTAP has prepared decision-point review documents for the Safety Function Status Trees that document the basis for technical data and numerical values used in the EOPs. The documents have been reviewed and approved by Reactor Safety Research, Reactor Operations Department, and Reactor Technology Department. The documents are not identified by a specific document number, but are part of the procedure package for DPSOL-105-MC-29-K, Rev. 2, "Vital Safety Function Monitoring-Emergency Condition" (Reference 3), presented for review and approval by the Operations and Maintenance Review and Approval Committee (OMPRAC). The documents become part of the procedure history file and are formally documented. Changes to the documents will not be tracked and reviewed since the documents are for historical purposes only. Administrative controls are required to ensure that all values in the EOPs are adequately maintained and revised as necessary. The staff determined that the EOP-decision point review documents provide the basis for the EOPs, but lack the

proper controls to ensure engineering data used in the EOPs is properly maintained. Pending the implementation of proper controls, this is an open item.

EOPs are to be written following a prescribed format and style. WSRC-IM-90-121, "FRP Writer's Guide" (Reference 4), which was approved on November 12, 1990, describes the format and style for the EOPs. DPST 85-1090, Rev. 9, "Procedure Writer's Guide" (Reference 5), describes the style and format for ACCs. DMP 7.01, Rev. 0, "Development and Control of Reactor Operations, Component Handling, and Maintenance Procedures" (Reference 6), defines the development and control processes for all Reactor Operations procedures. DMP 7.02, Rev. 1, "Operations and Maintenance Procedure Review and Approval Committee" (Reference 7), defines the review and approval process for all Reactor Department procedures. These documents together satisfy the restart criteria for the development and control of the EOPs and ACCs.

DMP 7.03, Rev. 0, "Symptom Based Procedure Verification and Validation Procedure" (Reference 8), defines the verification and validation (V&V) program for the EOPs. This procedure defines the purpose of procedure verification and validation as ensuring that the procedure is both written correctly (including human factors considerations) and technically accurate and that the procedure is operationally correct and useable by the operators. While procedure verification is primarily a table-top exercise, DMP-7.03 outlines three methods of performing the validation portion of the V&V process depending on the scope of the validation being performed. These methods are: (1) table-top, (2) walkthrough, and (3) simulator validation. Although the verification and validation process is proceduralized, the procedure does not address a time limit for steps that must be performed in the first 10 minutes of an event. Pending revision to DMP-7.03 to require verification of the operator's ability to perform the necessary actions within the specified 10 minute time frame, this is an open item.

DMP 7.01 establishes the requirements and responsibilities for the preparation, review, revision, change, and control of Reactor Operations procedures (of which the ACCs and EOPs are part). DMP 7.02 specifies that the review board has the responsibility for review and approval of Emergency and Abnormal Operating Procedures.

The staff has determined that it is not necessary to issue symptom-based status trees when adequate event-based Abnormal Condition Control (ACC) procedures are in place in the control room. This is due to the significant amount of development and operator training required prior to issuing EOPs. This waiver is, however, conditional on WSRC demonstrating that the event-based ACCs are adequate for response to accident conditions. Reactor Training and Procedures (RTAP) has endeavored to evaluate the acceptability of the ACCs by the use of the simulator. Event-based ACCs that are delineated in SAR Chapter 15 Accident Condition Categories II, III, and IV, with the exception of the Loss of All AC Power procedure, were reviewed and simulator malfunction codes were identified for each accident that could be performed on the simulator. The ACC procedures required for use in responding to each accident were evaluated by operators who respond to each of these accident scenarios. Only those ACC procedures necessary for response to specific

accident scenarios were evaluated. The revisions to the ACCs (required as a result of these evaluations) have not yet been approved nor have they been retested after revision. The acceptability of the event-based procedures is, therefore, still pending. Pending completion of this program to demonstrate the adequacy of the ACCs in response to accident conditions, this is an open item.

The adequacy of the ACCs that are not being performed on the simulator has been validated by field walkdowns by Technical Baseline personnel. These personnel do not have the plant-specific expertise necessary to ensure the operability of the procedures. In November 1990, five Reactor Operations personnel were assigned to RTAP to perform walkdowns. Pending the verification of the adequacy of the ACC walkdowns, this is an open item.

10.4.4 Conclusion

The staff has identified a number of open items with respect to the EOP program. Principal areas of concern are the Verification and Validation process, documentation, and administrative controls. Upon satisfactory resolution of those deficiencies, the staff will be able to conclude that the EOP program is acceptable for restart. The resolution of these open items will be addressed in a supplement to the SER.

10.4.5 References

1. WSRC, OPS-ROTD-89-0486, "Development and Implementation of Symptom Based Procedures for SRS."
2. WSRC, DPST-89-291, "Vital Safety Function Status Trees."
3. WSRC, DPSOL 105-MC-29-K, Rev. 2, "Vital Safety Function Monitoring - Emergency Condition."
4. WSRC, WSRC-IM-90-121, "FRP Writer's Guide."
5. WSRC, Manual RD 7.1, Procedure DPSP 85-1090, Rev. 9, "Procedure Writer's Guide."
6. WSRC, Manual RD 7.1, Procedure DMP 7.01, Rev. 0, "Development and Control of Reactor, Components Handling, and Maintenance Procedures."
7. WSRC, Manual RD 7.1, Procedure DMP 7.02, Rev. 1, "Operations and Maintenance Procedure Review and Approval Committee."
8. WSRC, Manual RD 7.1, Procedure DMP 7.03, Rev.0, "Symptom-Based Procedure Verification and Validation."

10.5 Housekeeping

10.5.1 Background

The appearance and proper functioning of station facilities, systems, and equipment are key indicators of a well-maintained, well-operated nuclear facility. Properly used, a housekeeping inspection program is an effective means of identifying and correcting station discrepancies. Maintaining good material conditions and ensuring adequate housekeeping inspections provide easier access for operations and maintenance personnel by reducing the number of potential radioactive sources and limiting the possibility that radioactive contamination may be carried outside of controlled areas. An adequate housekeeping program also aids in the identification of leaks or other potential problems.

The Reactor Operations Department (ROD) at the Savannah River Site (SRS) is the custodian of the reactor. In the past, the ROD was viewed as the only group responsible for housekeeping at the SRS reactors, and housekeeping standards were not effectively enforced. Due to the lack of emphasis on housekeeping by other groups performing work at the reactors and a relaxation in housekeeping standards, the SRS housekeeping program has been deficient.

In addition to ensuring that good housekeeping standards are set and communicated, a housekeeping program should include training for those involved in inspection activities, clearly defined inspection procedures, and a methodology for the documentation and followup of deficiencies. Followup of deficiencies to ensure that they are tracked through corrective action/disposition is also an important element of an acceptable housekeeping program. In addition, followup of corrective action following inspections is necessary for evaluating the timeliness and effectiveness of the corrective action and obtaining the maximum benefit from the inspection program.

WSRC is committed to revising the SRS housekeeping program to conform to Institute of Nuclear Power Operations (INPO) 85-038, Guidelines for the Conduct of Maintenance at Nuclear Power Stations, Chapter 13, "Station Material Condition Inspection" (Reference 1), including: (1) standards for housekeeping, (2) training of inspection personnel, (3) administrative procedures for program implementation, (4) scope of inspections, (5) size and frequency of inspections, (6) rotation of inspectors, and (7) reporting and follow-up of deficiencies.

10.5.2 Restart Criteria

The criteria for determining the adequacy of the SRS housekeeping program for restart are described below.

(1) Standards

A formal housekeeping program shall be developed and implemented using the guidance of INPO-87-038, Chapter 13, "Station Material Condition Inspection" (Reference 2). Setting good material and housekeeping standards involves establishing an atmosphere of proper work ethics, positive attitudes, and

specific expectations by management that are realistic, within the capabilities of the staff, and consistent with sound engineering judgement and good economic practice. Standards shall be communicated effectively to all personnel so that they are clearly understood. Adherence to these standards shall be assessed by plant managers and supervisors through the conduct of routine inspections.

Some indicators of good material condition and housekeeping standards are as follows:

- o Equipment is properly serviced (e. g., lubrication, drive belts, filters).
- o Fluid system integrity is maintained.
- o Protective cabinet doors and electrical enclosure covers are installed to maintain design integrity.
- o Plant equipment and systems subject to corrosion are protected with a preservative to minimize corrosion.
- o Industrial safety and radiological hazards are minimized (e.g., fire barriers maintained; trip hazards non-existent; radiological postings current and in place; radiological barriers and step-off pads properly established; sources of contamination identified).
- o Walkway and equipment access is maintained.
- o Equipment is clean (i.e., chemicals, oils, and solvents properly stored).
- o Plant areas, rooms, and grounds are maintained in a clean and orderly condition, including the proper storage of tools and material.
- o Coatings or coverings used to seal walls and floors in potentially contaminated areas are in good condition and assist in the control of contamination.
- o Illumination of areas, rooms, and grounds is maintained in a manner that provides sufficient light to perform inspections and minor maintenance.
- o Drums and containers are properly labeled.
- o First-level supervisors sign work orders indicating that a system and area have been cleaned and restored to their required status (for example, repainting where necessary).

(2) Training

Plant personnel involved in inspection activities shall be aware of standards established by plant management and the techniques required to perform material condition inspections. In addition, all personnel shall be aware of the importance of good material condition and housekeeping.

(3) Inspections

- (a) An administrative procedure shall be developed that describes the inspection program. This procedure shall: (1) define expected standards, (2) provide for documentation of deficiencies, (3) provide for a means to follow up on corrective actions for deficiencies, (4) assign responsibilities for program implementation, and (5) establish a means to measure program effectiveness.
- (b) Deficiencies identified during the inspection shall be documented by the inspector. Checklists indicating the equipment to be inspected and the types of deficiencies to look for shall be used as guides for the inspectors.
- (c) Inspections shall include detailed walkdowns of assigned areas, including remote and limited-access areas. The as-low-as-reasonably-achievable (ALARA) concept shall be considered when assigning inspectors and inspection frequencies for radiation zones.
- (d) Key individuals shall accompany the managers and supervisors during inspections. Discussions during an inspection should improve inspection techniques and convey the expected standards for plant material condition and housekeeping. The plant manager may designate specific types of deficiencies that should be given special attention during the inspection based on his/her desire to emphasize and upgrade particular aspects of material condition and housekeeping.
- (e) Routine inspections shall assess the following elements:
 - o Inspection areas shall be small enough to be thoroughly inspected in the time allotted.
 - o Periodic inspections of each plant area shall be scheduled.
 - o Inspectors shall be rotated through the various inspection areas periodically.
 - o Inspectors shall report inspection results to the inspection coordinator. Any significant material condition and/or safety deficiency shall be reported to supervisory personnel immediately for appropriate near-term attention.

- o All plant personnel shall be responsible for the prompt identification, correction (if feasible), or documentation of material condition and housekeeping deficiencies during the normal course of their duties.

(4) Deficiency Followup

- (a) A list of all deficiencies shall be prepared. This list shall include the names of personnel responsible for corrective action/disposition of the deficiencies.
- (b) Personnel responsible for corrective action/disposition shall report results of actions planned or conducted to the inspection coordinator periodically.
- (c) Deficiencies shall be tracked to completion in a timely manner.
- (d) Reported deficiencies shall be monitored to identify recurring, generic, and long-term problems. Action taken to resolve these problems shall include a failure analysis (not merely corrective symptoms). Followup of selected corrective actions from previous inspections is necessary for evaluating the timeliness and effectiveness of the corrective actions and for obtaining the maximum benefit from the inspection program. Several methods may be used to accomplish this followup. For example, the inspection coordinator may select corrected actions or those that are being corrected in a timely manner. This could be accomplished during each material condition and housekeeping inspection.
- (e) The inspection coordinator (or other assigned individual) shall periodically review inspection reports and actual plant conditions to evaluate the effectiveness of the inspection program. Inspection emphasis should be adjusted, where needed. The review shall also determine if changes should be made to any plant program to reduce generic, recurring, or continuing deficiencies.

10.5.3 Evaluation

WSRC is committed to implementing a housekeeping program that complies with the guidelines specified in INPO 85-038. In order to satisfy these criteria, WSRC issued DPSOP-403, "WSRC Housekeeping Manual" (Reference 3). This manual implements standards that establish an atmosphere of proper work ethics, positive attitudes, and specific management expectations. This manual will be superseded by RDP 3.02, "Housekeeping/Material Condition Control Procedure" (Reference 4), which implements the same standards established by DPSOP-403.

Assignment of responsibility for implementing the program is designated in RDP 3.02 for the Plant Manager and for the Maintenance Managers. The procedure specifically addresses program evaluation by the Housekeeping Coordinator, Plant Manager, and senior management.

The housekeeping standards describing the minimum acceptable criteria are fully contained in RDP 3.02 for the general areas of housekeeping, material condition, radiological controls, and safety. The standards incorporated into RDP 3.02 are discussed in the text of the procedure and are reiterated in checklists included as part of the procedure. These checklists are used by the inspectors during performance of the inspections. These housekeeping standards are very detailed and include specific items that fall under the following general headings: station cleanliness, walkway and equipment access, illumination of areas, industrial hazards, coatings and coverings, equipment cleanliness, proper operation of rotating equipment, properly serviced equipment, fluid system integrity, instrumentation operability, electrical equipment operability, electrical enclosures, unauthorized modifications, temporary equipment protection, equipment insulation, corrosion protection, and verification of complete area inspection.

The schedule used to specify the frequency at which the housekeeping inspections by various personnel are to be conducted is required to be developed by the Housekeeping/Material Condition Control Procedure. The housekeeping inspection schedule included in RDP 3.02 lists the individual inspection zones and provides for assignment of personnel to perform weekly inspections. The procedure requires that inspectors be rotated on the schedule such that each area is inspected by an inspector from a different organization each week. The inspection areas designated in RDP 3.02 are of limited size to permit a thorough inspection to be performed in the allotted time. Areas which are normally inaccessible during operation, such as the Process Water (PW) pump rooms, purification cells, etc. are also included in the inspection program. This procedure also contains a list of assigned inspection areas.

Inspectors within each shift/group are rotated as required to prevent familiarity with areas, which may hinder identifying deficiencies. The Plant Manager is required in RDP 3.02 to participate in periodic plant walkdowns with the staff members to ensure continuity of the housekeeping standards.

Deficiencies noted during housekeeping inspections are recorded on the Housekeeping Inspection Data Sheet of the procedure and forwarded to the Housekeeping Coordinator in accordance with the requirements set forth in RDP 3.02. Follow-up methods that track and trend repeat deficiencies are described in detail in this procedure.

The Housekeeping/Material Condition Control Procedure states that all personnel who work in the reactor facilities have a responsibility for good housekeeping practices and are to participate in housekeeping inspections, corrective action, and general housekeeping practices. Immediate reporting of significant material condition or safety deficiencies to plant management when found during an inspection or during the normal course of duties is required by the procedure.

The Housekeeping Coordinator maintains a record of findings and provides periodic reports to management on the status of deficiencies and past due items. The Housekeeping Coordinator also assigns groups responsible for taking corrective action and tracks the deficiency to completion in a timely

manner. Work groups are also required to provide updates on their assigned tasks, including outstanding housekeeping deficiencies, to the Housekeeping Coordinator.

The Housekeeping Coordinator is required to issue a quarterly report to all responsible organizations summarizing the deficiencies. A meeting must be scheduled with the Plant Manager to address recurring significant deficiencies, their root cause, and changes to the program to correct the problem. In addition, the Housekeeping Coordinator is to provide recommendations to the Maintenance Manager for periodic reviews and upgrades of the housekeeping program, conduct quarterly meetings with the Plant Manager to evaluate program effectiveness, and arrange semi-annual tours to assess overall program effectiveness.

RDP 3.02 requires training for personnel assigned to the Housekeeping Inspector be performed to ensure understanding of the requirements of the procedure, the minimum level of acceptable cleanliness, and the material condition of zones to be inspected. In addition, the procedure requires that all station personnel receive training on the general requirements of the housekeeping program and their responsibilities for maintaining general cleanliness and material condition of the areas. The staff determined that housekeeping training has been identified and included in operator restart training as a section in Training Cycle 3, Module K-AREA-006-LP-CT, "Formality of Work Practices." This training is also being given to plant support personnel as part of their reactor restart training. The DOE staff will evaluate the adequacy of this training as part of SER Section 10.2 "Training."

RDP 3.02 contains requirements that satisfy the restart criteria. However, RDP 3.02 has not yet been approved. Pending approval of this procedure, this issue will be carried as an open item.

10.5.4 Conclusion

The staff has concluded that the Housekeeping/Material Condition Control Procedure and Work Control Manual now in place, together with the training programs reviewed, will provide an adequate means of controlling station material condition and meet the restart requirements of the SER. However, as indicated above, the controlling procedure for this program is not yet approved. The closure of this open item will be documented in a supplement to the SER.

10.5.5 References

1. Institute for Nuclear Power Operations, INPO 85-038, "Guidelines for the Conduct of Maintenance at Nuclear Power Stations," Revision 01, December 1988.
2. _____, INPO 87-038, "Station Material Condition Inspection," Chapter 13.
3. WSRC, DPSOP-403, WSRC Housekeeping Manual, Revision 0, July 1989.

4. WSRC, RDP 3.02, "Housekeeping/Material Condition Control Procedure" (draft), January 2, 1991.

10.6 Logkeeping

10.6.1 Background

To provide an accurate history of plant operation, it is important to establish facility operating logs that fully record all necessary event data for all key shift positions. Information should be recorded in these logs in a timely fashion to ensure that log entries are accurate. In addition, management should clearly define the scope, type, and format for log entries to ensure that the operating information required by plant management is properly entered into the logs. All facility operating logs should document actions taken, activities completed, information transferred among operators, and data required for event reconstruction.

Deficiencies in the SRS logs were identified by several investigative groups (e.g., the Crawford Committee). These groups determined that WSRC's Reactor Operations Department (ROD) shift operating logs were inconsistent in scope, type, and format and that documentation of actions taken, activities completed, facility status, and the chronological sequence of events was not sufficient to provide the data necessary for an adequate and timely reconstruction and analysis of events. When a large, pressure transient (water hammer) event took place on January 22, 1989, for example, it was found that the logs did not contain sufficient information on plant status, activities in progress, or activities completed for the operator to determine the exact status of the plant. This lack of information contributed to the water hammer event.

10.6.2 Restart Criteria

The criteria for determining the adequacy of the SRS logkeeping program for restart are described below.

- (1) A formal logkeeping program shall be developed and implemented using the guidance of Chapter 11, "Logkeeping," of INPO 85-017, Guidelines for the Conduct of Operations at Nuclear Power Stations (Reference 1). Official narrative logs shall be established at all key shift positions in the facility. As a minimum, a narrative log shall be maintained by the Shift Manager or the reactor's Control Room Supervisor. For those shift positions that are manned on a part-time basis, a log should be established to ensure pertinent information is passed from operator to operator. A narrative section shall be provided on round sheets when a narrative log is not used for a particular shift position.
 - (a) Information shall be promptly recorded in the logs since delaying the recording of activities or events often leads to incomplete or inaccurate entries.
 - (b) The Reactor Operations Department management shall provide written guidance to thoroughly define the type and scope of entries for each log and the format for making entries. The following information shall be recorded in at least one facility log (although any one log might not contain all of these items).

- o reactor mode or condition (e.g., cold shutdown, power operation, run, start-up, refueling, etc.) and any changes that occur
 - o criticalities and appropriate critical data
 - o abnormal plant configurations
 - o status changes to safety-related and other major facility equipment
 - o occurrence of any reportable events
 - o initiation and completion of surveillance tests
 - o entry into and exit from any technical specification action statements
 - o security incidents
 - o out-of-specification chemistry results or indications
 - o shift reliefs
- (c) As much significant information as possible shall be logged during emergencies and abnormal or unexpected events to aid in event reconstruction. However, logkeeping shall not take precedence over controlling and monitoring the plant.
- (2) Log entries shall be made in permanent black ink such that they can be easily read, understood, and readily reproduced using standard photocopy machines.
- (3) Log entries shall not be erased or covered up. Corrections shall be made by placing a single line through the incorrect entry and writing in the correct entry in a nearby space. Corrections shall be initialed and dated.
- (4) Control room logs shall be reviewed and initialed by the Shift Manager, and logs kept by operators outside of the control room shall be reviewed by the Control Room Supervisor in charge of the reactor or an appropriate supervisor. These reviews normally would occur shortly before shift relief and shall ensure that entries are accurate and adequate. Additionally, the contractor's Operations Manager shall review the control room logs on a daily basis when in the area.
- (5) Management shall provide written guidance on the disposition of completed logs. This guidance shall address the following:
- o making the logs available for operators who return after periods of absence from their regular duties

- o storing the completed logs to ensure preservation for the expected life of the facility
- o retrieving stored logs should it become necessary

10.6.3 Evaluation

One of the supporting documents reviewed was Reactor Restart Review Transmittal P090-001, Issue Closure Package, MS-1, "Logkeeping" (Reference 2). This notice of closure pertains specifically to the requirements of the Charlotte Criteria. The notice of closure states that WSRC has (1) issued a revision to ROSI II.1, "Shift Operating Logs"; (2) implemented administrative controls governing logkeeping practices; (3) trained all of the operating personnel assigned on shift in K-Area on the required logkeeping practices; and (4) conducted an independent review of all area logs for conformance to the revised policy for all five operating shifts. Revision 4 to ROSI II.1 resolved deviations identified by this review.

ROSI II.1, "Shift Operating Logs," (Reference 3) has been issued to ensure shift operating logs will be maintained to record important events in reactor operations, information for exchange at shift turnover, and information for future reference and review. Narrative logs for key shift positions are established for the shift manager, components handling supervisor - C&D machines, control room supervisor, building supervisor, and assembly/disassembly supervisor for the recording of reactor operations information. In addition, the ROSI contains guidance on maintenance of logbooks for the remaining plant shift positions, including those manned on a part-time basis. The ROSI identifies the remaining shift positions requiring these logbooks and describes the minimum information required to be kept in these logbooks. The Conduct of Operations Manual also requires that control room and building operators maintain a narrative log. With this requirement, round sheets are no longer required to contain a narrative section for narrative-type entries.

ROSI II.1 provides written guidance to thoroughly define the type and scope of entries for each log and the format for making entries. The logs for the shift manager and components handling supervisor - C&D machines are required to contain the following minimum applicable information: (1) status of operations, (2) compilation of major items from all other logs, (3) C&D operations in the C&D log, (4) changes in the status of critical equipment, safety hazards, or radiation hazards, (5) damage or malfunctions of any major equipment, process system, or building structure, (6) sequence of events of what was accomplished by the shift, (7) accomplishments of required tests, inspections, and periodics, (8) any significant equipment removed from service for maintenance or repair, retests, and return to service, (9) actual versus scheduled outage work, (10) times of obtaining permission to start up or shut down, (11) safety and housekeeping activities, (12) deviations in operation that could result in a limiting condition for operation as specified in Technical Specifications/Technical Standards, (13) personnel accidents/injuries, (14) major work outages, (15) occurrence of significant events such as reactor scram, anomalous operations, unexpected power changes, (16) items deemed necessary to provide adequate information to the following

shifts and management, (17) events requiring notification and who was notified, and (18) additional information required by the plant manager.

In addition, the other shift positions are required to log the following minimum information: (1) significant events in the judgement of the operating personnel, (2) changes in the status of critical equipment or damage, (3) out of specification chemistry results, (4) safety and housekeeping results, (5) hydraulic operations activities, (6) purification operations, (7) building operations, and (8) C & D operations.

ROSI II.1 provides detailed logkeeping practices that are to be observed. Normally information is to be entered directly into the log as soon as possible. Logs are to be kept current with clear, concise, complete, and legible entries. All entries are required to be made in black ink. Errors in log entries are required to be voided by drawing a single line through the entry, initialled and dated by the person making the correction.

ROSI III.3, "Shift Assignment/Turnover" (Reference 4) states that each individual is responsible for reviewing and understanding the logs and data sheets applicable to the shift position. As soon as possible after shift turnover has been made, the relieving supervisor is required by ROSI II.1 to review all log entries made since his crew worked a shift. The Operations Manager is required to review the shift manager's logbook daily when in the area. During C&D operations, the components handling area manager is required by ROSI II.1 to review the C&D machines supervisor and assembly/disassembly supervisor logbooks daily. In accordance with the Conduct of Operations Manual, the building supervisor is required to review the building operator round sheets each shift. In addition, the shift manager is required to perform periodic reviews of the CCR and building logs to ensure that proper sensitivity is devoted to important items of logkeeping practices.

ROSI II.1 states that logbooks are legal records and must be maintained on file for the life of the plant. The shift clerk is assigned responsibility for maintaining and controlling completed logbooks in the CCR file in accordance with QA requirements. This file will then be available for operators who return from periods of absence from their regular duties. The Operations Manager is required to approve the removal of logs from storage.

Logkeeping practices were taught during Restart Training, Cycle 88-1, Training Module K Area-001-1G-CT, Rev. 0. This training was taught as part of the training on "Formality of Operations" given to Operations personnel. The assessments on the adequacy of training will be discussed as part of SER section 10.2, "Training."

10.6.4 Conclusion

The logkeeping program review is now complete. The DOE staff has determined that the logkeeping program satisfactorily addresses the restart criteria specified in the SER.

10.6.5 References

1. Institute for Nuclear Power Operations, INPO 85-017, Guidelines for the Conduct of Operations at Nuclear Power Stations, Chapter XI, "Logkeeping," Rev. 01, April 1988.
2. WSRC, Reactor Restart Review Transmittal P090-001, Issue Closure Package, MS-1, "Logkeeping," January 1990.
3. WSRC, Reactor Operations Standing Instruction ROSI II.1, "Shift Operating Logs," Rev. 5, October 1990.
4. WSRC, ROSI III.3, "Shift Assignment/Turnover," Rev. 8, November 1990.

10.7 Shift Turnover

10.7.1 Background

Shift turnovers are a critical part of reactor plant operation. During shift turnover, offgoing shift personnel are to provide oncoming shift personnel with information that reflects an accurate picture of the overall facility status (e.g., plant power level, reactor temperature, rod positions, abnormal conditions, planned maintenance, etc.). The Nuclear Regulatory Commission (NRC) identified inadequate shift turnover practices as an underlying contributor to the Three Mile Island (TMI) incident. Following TMI, the development and implementation of a formal shift turnover program became a requirement for all commercial operating reactors.

Deficiencies in the shift turnover process at the Savannah River Site were identified by both the Crawford Committee and Sandia National Laboratories. Shift turnovers at SRS were ineffective and did not provide oncoming operators with an accurate picture of overall facility status. Information provided during shift turnovers was also inaccurate and incomplete, and these inadequacies contributed to events at the facility. During the large pressure transient (water hammer) event that took place on January 22, 1989, for example, oncoming operators were given inadequate information, which contributed to the event.

10.7.2 Restart Criteria

The criteria for determining the adequacy of the SRS shift turnover program for restart are described below.

- (1) A formal shift turnover program shall be developed and implemented using the guidance of INPO 85-017 (Reference 1). The shift turnover program shall ensure that oncoming supervisors and operators are provided with an accurate picture of overall facility status.
- (2) Personnel shall not assume operational duties unless they are physically and mentally fit to do so and until they and the offgoing personnel have a high degree of confidence that an appropriate information transfer has taken place.
- (3) The shift turnover program shall require oncoming personnel to conduct a comprehensive review of appropriate written (logs, records) and visual (equipment, control boards) information before responsibility for the shift position is transferred. This may require oncoming personnel to arrive early and offgoing personnel to stay late so that adequate review time is available.
- (4) The shift turnover process shall require that shift turnovers are guided by a checklist and include a thorough review of appropriate documents describing important aspects of facility status and an inspection of appropriate facility instrumentation. These reviews shall then be complemented by a discussion between the offgoing and oncoming operators.

- (5) The shift turnover process shall require the use of the checklists specified below. Individual checklist items shall be initialed when completed and the completed checklist shall be signed. Oncoming operators shall review documents specified on their checklists prior to assuming responsibility for their shift position. Checklists may be used to identify documents which may be reviewed during the shift.
- (a) Shift Turnover Checklists: Each shift station position shall have a turnover checklist to be used in the turnover process.
 - (b) Facility Equipment Operator Checklists (Reactor Building Operator): Facility equipment operator checklists, or other formal documents shall provide for noting major components status (e.g., operating pumps, trains in service, etc.), any abnormal lineups, valid alarms on all pertinent local control panels, and surveillances or evolutions planned or in progress. Provisions shall also be made for documenting that a review of abnormalities has been performed by the oncoming operator.
 - (c) Control Room Operator Checklist: Certified Operator Checklists or other documents reviewed at shift change or shift briefing shall provide for recording the following facility status information as applicable to SRS: (1) plant power level; (2) reactor temperature; (3) reactor pressure; (4) important tank levels; (5) rod positions; (6) technical specification action statements in effect; (7) other important parameters; (8) required chemistry sample times; (9) most recent chemistry sampling results; (10) maintenance, surveillances, tests, or evolutions in progress or planned; (11) problems experienced with plant equipment and major equipment out of service; (12) when specific planned actions are required; and (13) changes in radiological conditions.

Certified operator checklists shall require documenting review of certain facility status documents. These shall include: (1) the operator's own operating (narrative) log, (2) the tagout (clearance) log, (3) the temporary modification and jumper logs, (4) the out-of-service equipment log, (5) the technical specification action statement log, (6) the shift order books, and (7) required reading since the last shift. Each station turnover checklist shall be provided with enough space for the offgoing operator to list other important information his/her relief should have.

- (d) Shift Supervision Checklists (Control Room Supervisor/Shift Manager): Shift supervision checklists or other documents reviewed at shift change shall provide the following facility status information as applicable to SRS: (1) reactor status; (2) evolutions (completed/in progress/planned); (3) controlled key status; (4) abnormal lineups or conditions; (5) changes in radiological conditions; (6) surveillance tests planned or in progress, and (7) maintenance planned or in progress.

Shift supervision checklists shall require documenting review of certain facility status documents. These shall include: (1) the supervisor's own narrative log, (2) the technical specifications action statement log, (3) the facility equipment status checklist, (4) shift order books, and (5) required reading since the last shift.

- (6) Document review shall be as intensive as necessary for the oncoming operator to understand important history, present status, and planned events. Narrative log entries for the previous 24-hour period or since his/her last shift shall be reviewed. Facility status logs shall be reviewed so that the operator is familiar with all currently active entries, with an emphasis on what has changed since he/she last had the shift.
- (7) Walkdowns of appropriate control boards shall be conducted by each shift station operator. The purpose of a board walkdown is to determine facility status through observation of system lineups, switch positions, lighted annunciators, chart recorders, and status lights.

Oncoming shift supervisors shall walk down the main-control boards before, during, or shortly after shift turnover. Oncoming and offgoing control room operators shall walk down their main control boards together. Facility equipment operators shall review all pertinent local control boards, (e.g., diesel, makeup water treatment, and ventilation) during their tour early in the shift. At a time when facility conditions are stable, the offgoing operator shall explain all items noted on the turnover checklist, and the oncoming operator shall ask any questions he/she might have. When both operators are satisfied that the oncoming operator is fully cognizant of the facility conditions, the oncoming operator shall state that he/she is assuming responsibility for the shift position.

Crew briefing shall be conducted by the shift manager. The briefing shall include a review of the facility status, problems with equipment, and evolutions in progress or planned during the shift. In addition to shift operators, personnel from support groups should also attend.

Reliefs occurring during the shift (e.g., exchange of control room supervisory function between supervisors) shall have a turnover that ensures the oncoming person is at least as knowledgeable of the facility conditions as he/she would have been had a complete shift turnover process been conducted. For example, control board walkdowns and log reviews may or may not be necessary depending on the oncoming person's familiarity with the current facility conditions. The shift manager or control room supervisor shall be notified prior to the occurrence of any on-shift relief.

10.7.3 Evaluation

ROSI III.3 (Reference 2) implements a formal shift turnover program to ensure each operator and supervisor is provided the knowledge required to adequately

and safely discharge the responsibilities of the shift position or assignment. This program describes the philosophy behind shift turnover as being an accurate picture to the oncoming supervisor or operator of the facility's status prior to assuming the responsibility for the shift position. ROSI III.3 provides directions and policies for the use and review of checksheets for CCR personnel, components handling personnel, and building personnel as a guideline for conducting shift turnover. Prior to assuming the shift position, oncoming operators are required by the ROSI to review pertinent information from data sheets, special instructions, and the applicable shift turnover checklists generated since their last shift rotation.

The CCR shift turnover checksheet identifies a number of items and conditions appropriate for shift turnover, including system status, any limiting condition for operation in effect, explanation of all abnormalities/alarms/annunciators, work in progress, critical parameters, logs and instructions to be reviewed, offgoing shift supervisor review, documentation of oncoming shift personnel reviews, changes in radiological condition status, and controlled key status. In addition, the turnover checksheet provides explanation and remarks sections to expand on any items identified on the checksheet or explain any item not identified on the checksheet.

The building supervisor and building operators have a turnover checklist that specifies items and conditions appropriate for shift turnover including system status, critical parameters, changes to radiological conditions, logs to be reviewed, and record of review and signature of offgoing and oncoming personnel. In addition, the turnover checksheet provides a remarks section to expand on any items identified on the checksheet or explain any items not identified on the checksheet.

In addition to the status provided on the turnover checksheets, these checksheets also contain a list of documents that need to be reviewed during the early part of the shift but not necessarily prior to assuming the shift. These documents include the shift manager's log, daily instructions, previous shift logs, data sheets, the equipment status log, the lockout log, the temporary modification log, the LCO tracking log, and the shift order book as appropriate to the shift position.

The SER restart criteria also requires all certified operators to review "required reading" since their last shift. "Required reading" is typically the name given to the program developed to keep operators abreast of those changes to the plant or procedures that are not deemed significant enough to address in either the shift orders or during shift briefings. In this case, however, the term is used to describe those changes that need to be brought to the attention of oncoming Operations shift personnel. Items deemed important enough to be covered during the course of shift turnovers are handled in one of two ways: described in the shift order book, which must be reviewed by all oncoming certified personnel; or discussed during the course of the shift briefing. In either case, items of this significance will be made known to oncoming personnel during the course of the shift turnover.

With respect to the turnover checklists, the DOE staff has determined that the ROSI does not meet the restart criteria in that the ROSI allows the operators to review all of the logs specified on the checklists after shift relief has occurred. The restart criteria allow the checklists to specify some logs to be reviewed after turnover, but allowing all log reviews to be handled this way does not meet the intent of the criteria. The DOE staff feels that as a minimum, the position narrative log, the LCO tracking log, and the equipment status log need to be reviewed prior to shift relief. Additionally, there is no way to document reviews of those logs that may be performed after shift turnover. Pending revision to the ROSI to require pre-turnover reviews (as discussed above) and to provide some method of documenting post-turnover reviews, this issue remains an open item.

ROSI III.3 requires shift operating personnel to conduct the shift turnover in a professional manner at all times. Each individual is responsible for reviewing and understanding the logs and data sheets applicable to his/her shift position before assuming the shift. No person is to permit his/her relief to assume the shift if there is doubt that the oncoming relief is alert, coherent, and fully capable of performing the assigned duties.

ROSI III.3 requires the oncoming shift personnel to read and sign the applicable shift turnover checksheets.

The DOE staff has determined that ROSI III.3 does not fully meet the restart criteria in that the ROSI checksheet does not require the oncoming operator initial for each individual completed item. Signing the completed checksheet does not, alone, fully ensure the completion of all items appropriate for shift turnover. Pending revision of ROSI III.3 to require initialling for individual checksheet items, this issue remains an open item.

ROSI III.3 requires the oncoming and offgoing supervisors/operators to perform a walkdown of the appropriate work stations prior to assuming the shift position. Discussions are to include as a minimum (1) status of safety-related systems, (2) running equipment and system alignments, (3) inoperable equipment and limiting conditions of operations, (4) reasons for alarms/annunciators, (5) locked-out equipment, (6) work in progress/scheduled, and (7) unusual conditions that have occurred since their last shift. Additionally, ROSI III.3 requires the oncoming shift manager to walk down the CCR work stations shortly after assuming the shift position.

ROSI III.3 requires the Shift Manager/Control Room Supervisor to conduct a shift briefing for all Operations personnel (and representatives of other area work groups as requested by shift management) to discuss items such as review of plant status, problems with plant equipment, evolutions planned, unusual conditions, and general shift routine. Normally, this briefing is done prior to turnover; however, it must be held no later than 1 hour after shift turnover. The CH Supervisor and Building Supervisor are required to conduct shift briefings with component handlers and building shift personnel, respectively.

ROSI III.3 states that reliefs occurring during the shift as a result of meetings, medical appointments, lunch breaks, etc. shall have a turnover that

ensures the oncoming person is at least as knowledgeable of the plant conditions as he/she would have been had a complete shift turnover process, in accordance with the ROSI, been conducted. The oncoming supervisor/operator shall notify the Central Control Room Operator upon relief of the regular supervisor/operator. The Central Control Room Operator is charged with notifying the Control Room Supervisor prior to on-shift relief occurring.

10.7.4 Conclusion

Open items in the shift turnover process are identified above. Upon satisfactory resolution of those open items, the staff will be able to conclude that the shift turnover requirements are sufficient for restart. The resolution of those open items will be addressed in a supplement to the SER.

10.7.5 References

1. Institute for Nuclear Power Operations, INPO 85-017, Guidelines for the Conduct of Operations at Nuclear Power Stations, Chapter 12, "Shift Turnover," Revision 01, April 1988.
2. WSRC, DPSOP-336, Reactor Operations Standing Instruction III.3, "Shift/Assignment Turnover," Revision 8, November, 1990.

10.8 Communications

10.8.1 Background

The purpose of the facility audible communications system is to alert plant personnel to normal, abnormal, or emergency conditions. Audible communication methods that are used to transmit operating and emergency information include horns, sirens, bells, telephones, radios, and the public address system. These communications must be controlled to ensure that they are available when needed and that they do not distract from plant operations during normal, abnormal and emergency conditions.

Accurate communications are essential to the safe and efficient operation of nuclear power facilities; however, information at the Savannah River site (SRS) has not been accurately transmitted within the facility. Informal transmission of operating and emergency information within the Reactor Operations Department (ROD) and between the ROD and other departments has adversely affected facility operations and contributed to the occurrence of events.

10.8.2 Restart Criteria

A formal communications program shall be developed and implemented using the guidance of Chapter 4, "Communications," of INPO 85-017, Guidelines for the Conduct of Operations at Nuclear Power (Reference 1). The criteria for determining the adequacy of the SRS communications system for restart are described below.

(1) Emergency Communications Systems

- (a) Methods shall be implemented to ensure all facility personnel are promptly alerted to facility emergencies. When personnel are working in areas where the public address system or emergency signals cannot be heard, alternate methods shall be devised for alerting these personnel. (Flashing lights, personal pagers that vibrate and can be felt, and individuals dedicated to notifications are examples of effective alternate methods.)
- (b) Emergency communications systems shall be tested periodically to ensure they are functional. Control room personnel shall have the capability of overriding other users of the public address system for emergency announcements.

(2) Public Address System

Use of the plant public address system (page) shall be administratively controlled to ensure that system effectiveness in contacting facility personnel is retained. Excessive use of the public address system for paging personnel or making unnecessary announcements shall be avoided. Excessive use of the public address system may reduce the impact of important announcements and can be distracting. Plant telephones and other point-to-point communications channels shall be used in lieu of

the public address system whenever practical. Consideration is to be given to dedicating certain paging system channels to specific groups or functions (e.g., a dedicated channel for routine operations or a dedicated channel used only for emergencies).

(3) Contacting Shift Operators

Methods shall be implemented to ensure that control room personnel can quickly contact on-shift operators or supervisors. The alternate methods cited in Restart Criterion 1, "Emergency Communications Systems," for communicating in high noise areas may be effective in accomplishing this function. However, to avoid operator confusion, distinction shall be made between a routine and an emergency notification.

(4) Radios

Portable radios can be an effective means of providing mobile point-to-point communications and may be used for this purpose. However, radio usage shall not be allowed in areas where electronic interference with facility equipment may result. Areas where radio use is prohibited shall be posted. Consideration is to be given to dedicating certain radio channels to specific groups or functions (for example, a dedicated channel for security or a dedicated channel while performing integrated surveillance tests).

(5) Abbreviations and Acronyms

Only abbreviations and acronyms obtained from an approved list (e.g., procedure-writers' guide) shall be used in facility communications. Both written and spoken terms should be prescribed in the list.

(6) Oral Instructions and Informational Communications

- (a) Oral instructions are to be clear and concise. In all communications, the sender and the intended receiver shall be readily identifiable. Instructions involving the operation of facility equipment should be repeated by the receiver to the extent necessary for the sender to ensure that the instructions are correctly understood.
- (b) Repeating instructions verbatim should be required by the controlling station when considered necessary to ensure accuracy of the order. Informational communications that do not require operator action need not be repeated; however, some acknowledgement that the message was received shall be given.
- (c) When transmitting alphanumeric information, phonetics should be used to ensure that letters are correctly received. For example, control room personnel may want the reactor operator to close a manual valve (e.g., PWS-002) in the process water system. The instructions might be transmitted over a communications channel

such as the telephone or radio in this manner: "Close Process Water Valve Zero Zero Two."

- (d) Although several standard phonetic alphabets are in use in other industries, using a standard alphabet is not as important as using some form of phonetics to transmit alphanumeric information correctly. For consistency, the phonetics appropriate for verbal communications are to be used to avoid misinterpretations.
- (e) The completion of directed actions shall be reported to the governing station, normally the control room.
- (f) Oral informational communications should be acknowledged so that the sender knows his/her message was received. Interdepartmental oral communications concerning facility operation, conditions, or status shall be coordinated through the control room. As a minimum, the control room should be informed of such communications. For example, a reactor operator might directly inform a radiological protection technician of adverse radiological conditions. This communication should be reported to the control room.
- (g) Written instructions or procedures shall be provided for complicated orders or other communications where important information might be forgotten. For example, extensive valving orders or extensive facility status reports should be written.

10.8.3 Evaluation

ROSI IV.8 (Reference 2) establishes guidelines for the reliable and accurate transmission of information on the operating shift and for uniform and enforceable policies for the control of wireless communications in the reactor building. The emergency action procedure identifies the type of emergency announcement to be made over the public address system for alerting plant personnel. The control room is required to have the capability for overriding other users (except security) of the public address system for emergency announcements. ROSI IV.8 states that all plant personnel must be promptly alerted to plant emergencies and that alternate methods should be devised for alerting personnel working in areas where the public address system or emergency signals cannot be heard. The ROSI specifies the use of vibrating pagers and flashing lights as alternate methods that shall be used to alert plant personnel to emergencies. WSRC has committed in the ROMP (item SE-15.0) to evaluate and upgrade the public address system to provide communication information in all normally occupied work areas.

ROSI IV.8 requires that the emergency communication systems be tested periodically to provide assurance of operability during an emergency. The ROSI outlines conditions under which the public address system may be used for communications. This use is restricted to situations like announcements of emergency conditions, announcements prior to significant activities, and plant personnel notification to the control room of fires, spills, injuries, or severe illnesses. The ROSI stresses that public address system use be kept to

a minimum to eliminate distractions and in no cases shall it be used in lieu of telephones or beepers to locate or communicate with individuals.

INPO 85-017, Chapter IV and the SER restart criteria pertaining to communications require methods to be implemented to ensure the control room can quickly contact on-shift operators or supervisors. ROSI IV.8 discusses the use of the public address system for the purpose of transmitting directions to operators if telephone or face-to-face communications are impractical. In addition, the ROSI provides guidance on how to announce emergency conditions so that there is no confusion between a routine notification and an emergency announcement.

ROSI IV.8 states that all forms of wireless communications equipment are forbidden in the reactor building except as specifically noted by procedure. Automatic exception is granted if wireless communication devices are required in the case of an area evacuation, during a complete power loss, or during a confirmed fire. Additionally, Wackenhut Services, Inc. (WSI) may use devices of this type during a security response. Under any other circumstances, their use can only be authorized if the following conditions exist: 1) no fissile assemblies and/or moderator are in the reactor tank, 2) the Emergency Cooling System is deactivated, and 3) a written request has been sent to, and approved by, the plant manager. DPSOL 105-1922-PLK, "Wireless Communication Device Control" (Reference 3), provides guidance for the shift manager to use in the control of this equipment.

ROSI IV.8 requires oral instructions to be clear and concise. In all communications, the sender and intended receiver are to be readily identifiable. All operating directions given verbally shall be acknowledged by the recipient. Their acknowledgement shall be accomplished by repeating back, or paraphrasing the directions, to ensure the instructions are understood. A list of acceptable abbreviations and acronyms is contained in DPSP-85-1090, "Procedure Writer's Guide", Appendix F (Reference 4). A suggested phonetic alphabet is provided in ROSI IV.8.

Frequently, orders to operators are lengthy or complex enough in nature that it is prudent for the operator to write down the instructions to ensure the actions are completed in their entirety and in the prescribed sequence. ROSI IV.8 specifies the need for written instructions or procedures for complicated orders or when important information might be forgotten. Additionally, the ROSI requires written instructions when the information being transmitted must be preserved for future reference.

After completion of directed actions, it is important that the operator report back that the directed actions have been completed. In addition to requiring this, the ROSI also directs the operator performing the action to report back the results of that action, if possible.

10.8.4 Conclusion

The Communications program review has been completed. The DOE staff has determined that the procedures now in place meet the restart criteria specified in the SER.

10.8.5 References

1. INPO 85-017, Guidelines for the Conduct of Operations at Nuclear Power Stations, Chapter IV, "Communications," Revision 01, April 1988.
2. DPSOP-336, "Reactor Operations Standing Instruction IV.8, "Communications," Revision 6, September 1990.
3. DPSOL 105-1922-PLK, "Wireless Communication Device Control."
4. DPSP-85-1090, "Procedure Writer's Guide", Revision 8, January 1990.

10.9 Tagouts/Lockouts

10.9.1 Background

Tagouts are used to control equipment removal from service and return to service. Tagouts also protect workers and equipment during maintenance activities. Errors in the tagout process must be prevented to ensure a high degree of personnel and equipment safety. In addition, it is necessary to correctly administer the tagout program so that the status and integrity of important plant components and systems are maintained.

Although the SRS tagout program provided adequate personnel protection, it did not provide sufficient control of equipment removal from service and return to service to assure the protection of the environment, facility components, and systems during operation, maintenance, and modification activities. A January 1989 Institute of Nuclear Power Operations (INPO) review revealed that the SRS tagout/lockout program was deficient to the point that control room supervision could not effectively determine plant status and evaluate the impact of tagging or equipment manipulations on the plant status. Also, the improper tagging of a check valve was a contributing cause to the large pressure transient (water hammer) event that took place on January 22, 1989.

10.9.2 Restart Criteria

The criteria for determining the adequacy of the SRS Tagout/Lockout program for restart are described below.

- (1) WSRC shall establish and implement procedures and standing instructions for tagouts. These guidelines shall require a tagout program equivalent to the commercial nuclear industry that implements the requirements of Chapter 9 of INPO 85-017, Guidelines for the Conduct of Operations at Nuclear Power Stations, Revision 1 (Reference 1).
- (2) The facility tagout program shall provide protection for personnel and facility components and systems during operation, maintenance, and modification activities. In addition, provisions within the tagout program shall provide for independent verification (in-process inspection) of the removal from service and restoration to service of safety-related or important-to-safety systems and for activities that have a potential for impacting worker safety.
- (3) Tagouts shall be used to control equipment removal from service and return to service and to protect workers and equipment during maintenance activities. The tagout program shall include:
 - detailed administrative procedures
 - uniquely identifiable tags
 - appropriate control over tagout preparation, approval, placement, and removal
 - adequate documentation
 - "Danger" (clearance, safety, out-of-service) tags that prohibit component operation

- "Caution" tags that alert operators to particular situations affecting certain components shall be provided
- (4) All aspects of the tagout program shall be defined in appropriate facility procedures. The procedures shall be understood and complied with by all personnel who work with the tagging system. The tagout procedure shall contain a logical sequence of necessary steps to establish a safe tagout. In addition, the tagout form described in the tagout procedure shall provide a logical sequence of events to follow in documenting the tagout evolution and guiding the process. Other elements that shall be contained in the tagout procedure include the following:
 - o clearly defined terms
 - o assigned duties and responsibilities that specify detailed action
 - o guidance for determining the boundary points, as well as vent and drain paths (where applicable), based upon the purpose for each tagout request and the nature of the potential hazard (e.g., electrical, temperature, pressure, chemical, or toxic substance)
 - o controls that ensure work does not begin until equipment positioning is complete and tags are in place
- (5) Tags used for personnel or equipment protection shall be uniquely identifiable and different in appearance from other facility tags. The meaning of the tags used for personnel or equipment protection shall be understood by all facility personnel. The following information shall be included on the tags to assist facility personnel in administering the tagout program:
 - o tagout and tag number
 - o component name and number/designation
 - o component position required for safety

Miniature tags shall be used in cases where the use of a full-size tag could obscure switches, indicators, or other control devices or information devices.

- (6) Tagouts shall be prepared by individuals qualified in tagout preparation. Identification of components to be tagged shall be based on controlled documents that identify all required boundary points (valves, circuit breakers, fuses, etc.). These documents shall also be used to identify all equipment that would be affected when boundary points are isolated. Normally, controlled facility drawings will provide the best documentation for tagout preparation. When appropriate, tagouts shall address both local and remote operators and include the sequence in which components are to be positioned and tagged. In general, the isolation sequence specified shall be from higher energy to lower energy components. Check valves and other passive components shall not be relied upon to protect piping, instrumentation, and personnel. Tagouts shall be prepared by individuals

qualified in the preparation of tagouts who are fully aware of the scope of work to be performed.

- (7) Tagouts affecting facility equipment shall be approved by the person who has the control room supervisory function for the affected reactor. Normally, this will be the shift manager; however it could be another person who is senior reactor operator certified. In this case, however, the shift manager shall approve tagging out of safety-related equipment. In addition, the shift manager or control room supervisor shall verify that the tagout does not violate technical specifications and shall arrange for any testing of redundant components or systems that is required to satisfy technical specifications.
- (8) A record of all active tagouts and the position of all tagged components shall be available to control room personnel. Control room personnel shall be able to use this documentation to determine the effect of the tagouts on the facility.
- (9) Tags shall be placed by operators qualified and knowledgeable of the particular components and systems to be tagged. For safety-related or important-to-safety systems and for activities that have a potential for impacting worker safety, tag installation and component position shall be independently verified by a second qualified individual. This independent verification is in addition to the verification performed by the worker or job supervisor mentioned below. Tags shall be placed so that it is obvious what component is affected by the tag, and so that anyone intending to operate the equipment is aware of the tagout. The person placing the tags and the person verifying tag placement shall individually document the placement or verification as each action is performed. Prior to the start of work, the worker or the job supervisor shall verify that the tags have been placed and that the boundary components are properly positioned.
- (10) The shift manager or other cognizant control room supervisor shall authorize the removal of tags. The tagout procedure shall specify a method for authorizing tag removal when the workers or their supervisor cannot be contacted. In this case, a thorough inspection of the job site by a supervisor shall be conducted, and a maintenance supervisor shall concur prior to the removal of tags. The individual authorizing tag removal shall specify the final component position and, when appropriate, the sequence in which components shall be positioned. In addition, the need to check the lineup of other components that were not tagged, such as valves inside the isolation boundary, shall be determined at this time and appropriate instructions developed. These checks shall be done, when necessary, to ensure that components within the tagout boundary are correctly aligned to support intended operation.
- (11) An operator qualified to operate the tagged equipment shall remove the tags and reposition the components in the sequence specified. The operator shall also check the lineup of any additional components as deemed necessary by the person who authorized the tag removal. The removal of the tags and the lineup of the components shall be

documented. Tags shall be returned to the shift manager or other cognizant control room supervisor. The supervisor shall satisfy himself/herself that all tags and safety devices, such as grounds, associated with this tagout are removed and accounted for.

- (12) Independent verification of systems or component alignment shall be performed and documented following removal of tags on safety-related and other important systems. These verifications shall include components that, if mispositioned, could cause reactor shutdown or challenges to safety systems. The plant manager shall specify those components that require independent verification either individually or by system. The systems requiring independent verification shall be specified in accordance with the restart criteria specified in section 10.12.
- (13) Tags may be temporarily removed for testing or other required brief periods of equipment operation. For example, it may be necessary to remove or replace a tag several times when balancing rotating machinery. Temporary removal shall not be allowed for extended periods. Normally, a temporary tag removal shall not extend beyond the next shift change. When tags are temporarily removed and placed, the same controls associated with normal removal and placement of tags shall be observed. For example, temporary removals and replacements shall be approved, documented, and verified in a manner similar to those actions for initial tag placement and final removal.
- (14) Tagouts shall be administratively reviewed at least monthly. This review verifies the continuing need for each active tagout and ensures the tagout index accurately reflects all active tagouts. A certified operator is ideal to perform the review because his/her knowledge of systems and equipment might be necessary to accurately evaluate the continuing need for a tagout. Input from the clearance tagout holder shall be solicited as needed. This review shall be documented.

An audit of randomly selected tags and tagouts shall be performed at least quarterly. All tags shall be checked at least annually, and a complete audit of all tags shall be conducted prior to start-up from extended outages. Audits shall include the following checks:

- o proper placement of each tag
- o proper position of tagged equipment
- o accuracy, completeness, and legibility of information on tags and tagout sheets

During the course of these audits, a tour of facility spaces shall be made to check for the presence of unauthorized tags, superseded tags, and tags for which no record exists. These tags shall be investigated and either removed (with concurrence of the individual by whom the component was tagged, if applicable) or authorized, whichever is appropriate.

Deficiencies noted in the audits shall be documented and corrective action shall be taken. If a significant number of deficiencies are found, the extent of the audit shall be increased to provide statistical assurance that the remaining tagouts are in order.

- (15) Caution tags shall not be used for personnel protection (i.e., caution tags shall never be used in place of a safety [clearance] tagout).

Administration of the caution tag program shall be accomplished as a part of the safety tagout program.

The use of caution tags shall be restricted to those situations in which a component or system is functional, but some precaution or information is necessary prior to operation.

Caution tags shall be uniquely identifiable and differ in appearance from other facility tags. The following information shall be included on the tags as an aid in administering the program:

- o caution tag number
- o component name and number
- o effective date
- o precaution or information applicable to the particular situation and/or component or system
- o signature of authorizing individual

Pre-issuance review: Situations that require special operator precautions or amplifying information shall be brought to the shift manager's attention. The shift manager shall ensure that issuing a caution tag is necessary and that the tag is not being used in place of more appropriate administrative action (e.g., a temporary procedure change, use of the work control system, or issuance of a safety tagout). The shift manager review shall also ensure that any instruction contained on the caution tag does not deviate from established facility procedures or technical specifications.

Caution Tag Records: A record of all active caution tags and the amplifying information contained on them shall be available to control room personnel. This record and associated tags shall be reviewed at least monthly by a certified operator. This review verifies the continued need and applicability for each caution tag and ensures the caution tag index accurately reflects all active caution tags. This review shall be documented. Any caution tags that must remain in an active status for extended periods of time (e.g., longer than 3 months) shall be brought to the attention of the plant manager. The plant manager shall determine what action needs to take place to resolve the continued use of the caution tag.

Caution Tag Placement: Caution tags shall be placed such that they do not interfere with or obscure indications, switches, or other control devices, but are readily apparent to an individual prior to the operation of the tagged device. Placement of caution tags shall be

documented either in the caution tag log or on a separate form that is maintained in the log.

Caution Tag Removal Authorization: The shift manager or the control room supervisor shall authorize the removal of caution tags.

10.9.3 Evaluation

WSRC Procedure RD 3.1 Appendix A "Lockout" (Reference 2) provides the means of writing instructions for the isolation and restoration of equipment and systems to ensure the safety of personnel and equipment during maintenance, modification, inspection, and testing activities. The process described in Appendix A is arranged in a logical manner. The general requirements are followed by the procedure steps for lockout requests, lockout preparation, review and approval, lockout authorization, lockout installation, lockout holder acceptance, lockout holder release, locking and tagging authority release, lockout modifications, and caution tags. The last section of RD 3.1, Appendix A describes administrative controls, including maintaining the required documentation and performance of lockout audits. The various supporting forms, such as the Lockout Request, Lockout Order, and Lockout Modification forms, are similarly arranged in order to guide and document the tagout process.

In order to ensure the terms used in the tagout process are clearly defined and available for easy reference, RD 3.1, Appendix A contains a list of the approved terms and their corresponding definitions.

RD 3.1 clearly assigns specific duties and responsibilities for the Locking and Tagging Authority, Locking and Tagging Operator, Lockout Holder, Plant Manager, and Reactor Engineering as related to the tagout/lockout program.

RD 3.1 assigns the Locking and Tagging Authority the responsibility for ensuring assigned personnel are properly qualified for the preparation and approval of tagouts by having successfully completed reactor area locking and tagging training.

The Lockout Planner is tasked with, and provided guidance for, determining proper isolation boundary points as well as vent and drain paths by RD 3.1. The procedure addresses the lockout of breakers within 110-volt electrical distribution panels, use of pulled fuses and grounding devices, and diaphragm air-operated valves. It also provides general guidance for use in the preparation of lockouts, including not using pushbuttons, relays, solenoids, and check valves as lockout points. The procedure specifies the sequence to be used in locking out a component. This sequence identifies motive power sources, control power sources, all control switches, and any local operating devices or control switches.

RD 3.1 requires the use of controlled document sources such as DPSOLs, DPSOPs, prints, drawings, and vendor manuals to ensure current and accurate information is used in the preparation of lockout orders. All references used in the development of lockout orders are required to be documented on the lockout order form.

In order that control room personnel be able to adequately determine the effect of the tagout on the facility, a lockout log and lockout order book are required to be maintained to provide a record of all active tagouts and the position of all tagged components.

Tags used for personnel or equipment protection are required to be uniquely identifiable and different in appearance from other facility tags. RD 3.1. Appendix A requires the Locking and Tagging Authority to enter specific information on Danger/Do Not Operate tags. Included in this information are a lockout number that is unique to all the tags in that particular lockout and a tag number. Each tag in a specific lockout is assigned a different tag number to provide a uniquely identifiable tag. Also included on each tag is the component name and number/designation and the component position to be maintained.

RD 3.1, Appendix A requires the Locking and Tagging Authority to review the Lockout Request to determine if the request can be implemented based on a number of considerations, including personnel and equipment protection. Other considerations are technical specifications and operational requirements or limitations, safety and fire protection requirements, high energy source isolation and release, draining of fluids, possible personnel hazards associated with performing the tagout, and conflicts with existing lockout boundaries.

RD 3.1, Appendix A requires Reactor Engineering evaluate the technical specification impact for all lockouts on equipment and systems important to safety. Before authorizing the installation of the lockout, the Locking and Tagging Authority is required by RD 3.1 Appendix A to notify all affected groups (e.g., security, fire department, health protection) that equipment will be declared inoperable and removed from service. He/she is also required to verify that plant conditions will support the proposed system alignments and that appropriate approvals have been obtained. The Locking and Tagging Authority authorizes installation of the lockout by signing on the Lockout Order.

RD 3.1 Appendix A provides guidance for both first and second operators performing the lockout. The first operator is required to (1) have the Lockout Order and instructions in his possession during lockout installation, modification, or removal; (2) ensure component labeling agrees with the lockout plan; (3) position components per the lockout; (4) place each tag such that it is clearly visible and does not interfere with or obscure indications, switches or other control devices; (5) sign and hang each tag; (6) and return the lockout order and instructions to the Locking and Tagging Authority. The second operator (1) performs independent verification for each lockout point per attachment 1 to RD 3.1; (2) installs locking devices per lockout order; (3) performs try-on installed locking device; (4) signs the tags; (5) deposits the keys in the supervisory lockbox; (6) locks, performs try, and tags the lockbox; and (7) returns the lockout and instructions to the Locking and Tagging Authority.

Tagout/lockout administrative controls are required to ensure work does not begin until equipment positioning is complete and the tags are in place. The

work group supervisor, or individual worker, is required to accept the lockout by signing on the Lockout Order as a Lockout Holder. This signature signifies acceptance of the lockout as properly installed and adequate to provide personnel safety during the work activity. RD 3.1 states that prior to authorizing work activities, the shift manager shall verify that the required equipment lockouts have been established and accepted. The shift manager indicates his approval by signature on the Reactor Work Request form.

Prior to authorizing release of the lockout, the Locking and Tagging Authority verifies that any components inside the lockout boundary that may have been manipulated have been restored to their required position. Lockout holders sign the release section of the Lockout Order when the equipment is ready for turnover to operations including completion of performance testing and restoration of the work area. If the lockout holder is unavailable, management may release the lockout. The Locking and Tagging Authority is responsible for authorizing the actual tag removal and for completing the removal sections of the Lockout Order, removing the lockout, and aligning the system. The Lockout Order contains blocks for removal sequence and final position for all tagged components. The shift manager is authorized to release a lockout in the event of an emergency that poses an immediate threat to reactor or personnel safety.

The Locking and Tagging Authority is required to assign an authorized locking and tagging operator to remove the lockout and return the uncontaminated tags to the work control supervisor. The operator documents the removal of each tag and the final position of each tagged component on the appropriate blocks on the Lockout Order. The Locking and Tagging Authority is required to account for all tags and then destroy them.

The Locking and Tagging Authority is required to assign another authorized locking and tagging operator to perform an independent verification of the lockout removal and alignment when the components affected require independent verification.

RD 3.1 Appendix A provides guidance for auditing the lockout program. A weekly audit of active lockouts (approximately 10%) is required to be performed. This audit checks the following items: (1) equipment is properly tagged and locked, (2) equipment is inspected for proper position, (3) any missing, damaged, or illegible tags are replaced, (4) no unauthorized tags are in place, and (5) documentation is in compliance with this procedure. If deficiencies are found in greater than 20 percent of those lockouts randomly sampled, the sample shall be expanded to include another 10 percent of all active lockouts. If deficiencies are found in greater than 20 percent of the expanded sample, a full audit is required to be performed. A monthly audit of the lockout log is required to ensure that the log accurately reflects all active lockouts and that all active lockouts are still needed. A quarterly audit of 100% of all active lockouts is required. The same items should be checked during these quarterly audits that are checked during the weekly audits. A 100% audit of all active lockouts shall be performed prior to reactor startup following an extended outage.

RD 3.1 Appendix A also contains guidance on the use of caution tags. The procedure states that caution tags should only be used to convey amplifying information. The person requesting the installation of the tag is required to completely fill out the tag, including the amplifying information. The Locking and Tagging Authority is required to authorize the installation of each caution tag. Caution tags are required to be hung on or near the affected equipment. A log of all installed caution tags is required to be maintained by the Locking and Tagging Authority. Caution tags are required to be audited in the same manner as danger tags.

In order to preclude the possibility of having conflicting guidance concerning the performance of tagouts in the reactor areas, revision 2 to RD 3.1 specifically states that the procedure is applicable to all personnel working in the reactor areas. This change ensures uniformity in the application of the requirements concerning tagouts.

10.9.4 Conclusion

The staff's review of the tagout/lockout program has been completed. The staff has determined that the program now in place meets the restart criteria specified in the SER.

10.9.5 References

1. Institute for Nuclear Power Operations, INPO 85-017, Guidelines for the Conduct of Operations at Nuclear Power Stations, Chapter 9, "Tagouts", Revision 01, April 1988.
2. WSRC, Reactor Division Procedure RD 3.1, "Work Control," Appendix A, "Lockout", Revision 2, January 1991.

10.10 Notifications/Reporting

10.10.1 Background

Events requiring the notification of offsite personnel and/or regulatory agencies can occur frequently at nuclear facilities. It is vital that information be gathered and transmitted in a consistent, systematic, and controlled method in accordance with DOE Orders and national security requirements. Improving the consistency and quality of event reporting is an essential element in strengthening the line management role in operations.

At the time of shutdown at SRS, procedures defining notification responsibilities and adequate documentation of abnormal events did not exist. Inconsistent and uneven reporting practices exist within DOE, with the result being that top DOE managers are too often unaware of actual conditions that exist at their facilities.

DOE Order 5000.3A, "Occurrence Reporting and Processing of Operations Information" (Reference 1) cancelled DOE Order 5000.3, "Unusual Occurrence Reporting System." This new order establishes a system for reporting of operations information related to DOE-owned facilities and processing of that information to provide for appropriate corrective action.

10.10.2 Restart Criteria

The criteria for determining the adequacy of the SRS Notifications/Reporting program for restart are described below.

(1) Notification Procedures

- (a) Procedures shall be developed to address the requirements for appropriate notification upon the declaration of any of the emergency action levels specified in the Westinghouse Savannah River Company (WSRC) Emergency Plan. This notification shall be no later than 2 hours after the time WSRC declares one of the emergency action levels.
- (b) Procedures shall be developed to address appropriate notifications in accordance with INPO 85-017, Chapter 6 and 7, (Reference 2) and the following DOE Orders and national statutory requirements:
 - DOE 5484.1 Chapter 1 (Reference 3)
 - DOE 5000.3 Part 9 & 10 (Reference 4)
 - DOE 5400.1 Chapter 2 (CERCLA) (Reference 5)
- (c) Procedures shall be developed to address appropriate notifications for events, conditions, or concerns that do not violate any specific DOE Order, but for which DOE should be informed to ensure open communication between WSRC and DOE. Although there is no required timeframe in which DOE must be notified of these off-normal occurrences, this notification should normally be done within 2 hours of classifying the off-normal occurrence. Examples

of these types of events can be found in Appendix 1 to this section.

(d) WSRC shall ensure that reactor area procedures fully implement the Site Item Reportability and Issue Management (SIRIM) System. The following are some specific elements that shall be included in these procedures:

- o specific responsibilities for notifications
- o identification of events and conditions requiring notification
- o identification of primary and alternate personnel to be notified for the different situations
- o establishment of time requirements for notifications consistent with the facilities emergency plan or regulatory guidance
- o definition of recordkeeping requirements that document the reason for notification, time of notification, and person notified

(2) Notification Responsibility

The Shift Manager, with the assistance of other individuals, (e.g., Staff Duty Officer, Site Spill Coordinator) is the person responsible for ensuring that all appropriate personnel receive notification. The actual notification of specific individuals or agencies may be accomplished by other individuals.

(3) Names and Phone Numbers

The names of primary and alternate contacts, current phone numbers, and pager codes shall be readily available to the personnel assigned to make notifications.

(4) Documentation

All notifications shall be documented and maintained using a formal record of notification.

(5) Communication Equipment

Adequate communication equipment shall be maintained in the Central Control Room and the Technical Support Center to meet the objectives of the notification procedure.

10.10.3 Evaluation

DOE Order 5000.3A lists the DOE complex-wide requirements for occurrence reporting and processing. INPO 85-017, Chapter VI, concerns the investigation of abnormal events. Areas described include Events Requiring Investigation, Investigation Responsibility, Investigator Qualifications, Information to be gathered, Event Investigation, Investigation Report, Event Training, Event Trending, and Sabotage. Chapter VII provides guidelines of Notification Procedures, Notification Responsibility, Names and Phone Numbers, Documentation, and Communication Equipment. At SRS, these requirements are addressed by Management Requirements and Procedures (MRP) 4.07 through 4.10 (References 6 through 9). MRP 4.07 specifies the responsibilities and requirements related to the identification and initial notification of plant management of an event. MRP 4.08 follows with the requirements for determination of actions to be taken in response to an event or condition. MRP 4.08 also contains the required callout list. MRP 4.09 provides the requirements for follow-up, tracking, and evaluation of events and concerns. MRP 4.10 defines the role of the WSRC Site Issues and Employee Concerns Coordinator after being informed of an event or condition requiring DOE notification and contains a required callout list. The requirements of MRP 4.07 through 4.10 are implemented in the reactor areas by RDP 12.02, "Notification of Reactor Plant Conditions, Events, and Concerns" (Reference 10). RDP 12.02 also incorporates practices which have been instituted as a result of operating experiences, lessons learned, and management directives.

RDP 12.02 specifies the conditions, events, and concerns that require notification by Reactor Operations Department personnel to onsite and offsite authorities. Notification begins with the identification of an event or concern that may have an effect on quality, health, safety, security, or the environment. An emergency will result in immediate performance of appropriate corrective actions per the operating or emergency procedure. Parallel actions include conducting offsite notifications in accordance with the 6Q7 manual Emergency Action Levels (EALs). After the emergency is under control, a critique is scheduled. Actions following and including the critique are addressed in the same manner as non-emergencies. Non-emergencies are classified by the Shift Manager criteria in RDP 12.02. Following the immediate notification of the CCR, the Shift Manager will take appropriate corrective actions in accordance with the applicable procedures. The Shift Manager then completes the 24-hour report, makes the required notifications, and documents them in the CCR Log. Notification is required within 2 hours. Following the notifications, a critique is scheduled and is held by the end of the shift.

RDP 12.02 describes the personnel responsibilities for notification. It states that the Shift Manager (Area Emergency Coordinator) is responsible for evaluating operational and other real-time conditions, events, and concerns for notification requirements. The Shift Manager is responsible for reviewing all written and verbal reports of conditions, events, and concerns for impact on operations and for notification requirements. The Shift Manager is also responsible for ensuring that initial, follow-up, and final notifications are made promptly and within the specified time limits and that conditions, events, and concerns requiring notification are properly documented. The RDP

states that the Operations Manager is responsible for ensuring up-to-date telephone directories of DOE representatives, WSRC personnel, and the Response Center are maintained in their respective control rooms. RDP 12.02 states that the managers of Reactor Quality Assurance and Assessments, and Reactor Engineering are to provide a current 24-hour call-out list of knowledgeable personnel designated to attend critiques of notifiable conditions, events, and concerns.

RDP 12.02 details the actions to be taken to identify and classify conditions, events, and concerns about which DOE representatives and WSRC management personnel must be notified. The classification is performed only after immediate action is taken to prevent or minimize personnel injury or equipment damage. RDP 12.02 contains a comprehensive flow chart to assist the Shift Manager in executing all the appropriate procedures and subsequent actions necessary to support the classification and notification process. The Shift Manager is responsible for classifying an emergency condition, event, or concern in accordance with RDP 12.02 and Procedure Manual 6Q7 "Reactor Area Emergency Plan and Emergency Plan Implementing Procedures." RDP 12.02 is structured to specifically handle the classification and notifications involved in Unusual Occurrences through the use of Attachment B, "Matrix of Notification Requirements For Conditions, Events and Concerns." This matrix provides a detailed listing of conditions, events, and concerns that the Shift Manager uses to classify issues and ensure proper immediate notification.

RDP 12.02 provides instructions about how the notifications are made. If the condition, event, or concern requiring notification is an environmental issue (i.e., a hazardous material accident, spill, release, or noncompliance) the Shift Manager notifies the site spill coordinator and the Operations Manager within 2 hours of the classification of the condition, event, or concern. The Shift Manager then ensures that the requirements for spill reporting and control are satisfied. The RDP provides the notification requirements if the condition, event or concern is not environmentally related. Notifications are also required to be within 2 hours of the classification.

Attachment B of RDP 12.02 specifically addresses event notification and documentation requirements. A 24-hour notification report (Attachment D) is included with the appropriate instructions to ensure that the report is correctly filled out (Attachment C).

RDP 12.02 describes how notifiable conditions, events, and concerns are documented. The Shift Manager performs follow-up notifications and documents their completion by completing Item 10 of the 24-hour Notification Report. If required, he/she is also responsible for initiating nonconformance reports (NCRs) or discrepancy reports (DRs) and event critiques. Event critiques are required for all Category I (EAL) and II (Regulatory Reportable) occurrences, and for selected Category III (Off-Normal) occurrences as determined by the Shift Manager. The report form and attachments (NCRs, DRs, and critiques, if any) are delivered to the Operations Manager for review. The Operations Manager is to complete the review and deliver the package to the Plant and Human Performance Assessments Manager within 24 hours of the classification.

The Plant and Human Performance Assessments Manager is responsible for evaluating the 24-hour notification report to determine if the event has been properly classified. If, in his/her opinion, the event requires reclassification, he/she notifies the Reactor Area Shift Manager, who processes the change in accordance with RDP 9.02, "Occurrence Reporting System" (Reference 11). The reporting process in RDP 9.02 begins with the identification of potential Unusual Occurrences (UO) by RRD employees. Notifications are then made to management. An investigator performs an investigation and prepares a Ten-Day Report. The completed Ten-Day Report is then approved and issued. Following the issuance of the Ten-Day Report, the root cause of the event is determined and the corrective actions required to prevent recurrence are developed and tracked. A final report is then prepared. Corrective actions which must be included in Strategic Programs are identified. RDP 9.02 establishes the responsibilities and defines the requirements for the identification, investigation, documentation, and approval of Occurrence Reports for the Reactor Restart Division.

DOE Order 5000.3A requires that the DOE Program Senior Official (PSO) approve procedures for implementation of the requirements of this Order. The WSRC implementing procedures have not yet been approved by the DOE PSO. This is an open item.

10.10.4 Conclusion

The DOE staff has reviewed the notification/reporting program and identified the above open item. The staff will report on the resolution of the open item in a supplement to the SER.

10.10.5 References

1. U. S. Department of Energy, DOE Order 5000.3A, "Occurrence Reporting and Processing of Operations Information," May 1990.
2. Institute for Nuclear Power Operations, INPO 85-017, Guidelines for the Conduct of Operations at Nuclear Power Stations, Chapter 6, "Investigation of Abnormal Events," and Chapter 7, "Notifications," Revision 01, April 1988.
3. U. S. Department of Energy, DOE Order 5484.1, "Environmental Protection, Safety, and Health Concerns."
4. _____, DOE Order 5000.3, "Unusual Occurrence Reporting System."
5. _____, DOE Order 5400.1, "General Environmental Protection Program."
6. WSRC, MRP 4.07, "Events, Conditions, and Concerns Identification," Rev. 0, August 1989.
7. WSRC, MRP 4.08, "Events, Conditions, and Concerns Determination," Rev. 0, August 1989.

8. WSRC, MRP 4.09, "Savannah River Site Issue Management," Rev. 0, August 1989.
9. WSRC, MRP 4.10, "Site Issues and Employee Concerns Coordinator," Rev. 0, August 1989.
10. WSRC, RD 16.1/RDP 12.02, "Notification of Reactor Plant Conditions, Events, and Concerns," Rev. 1, December 1990.
11. WSRC, RDP 9.02, "Occurrence Reporting System," Rev. 0, November 1990.

Appendix 1

Reportable Events Notification Requirements

<u>Type</u>	<u>Reference</u>	<u>Part</u>	<u>Type of Notification</u>	<u>Follow-Up Required</u>
1	DOE 5484.1	1.1	Immediate (4 hour) verbal to DOE-HQ	Written report in 48 hours in DOE 5000.3 format
2	DOE 5484.1	1.2	72 hour written to DOE-HQ	None
3	DOE 5000.3A	7	24-hour written to DOE-HQ	Occurrence Report
4	CERCLA	N/A	Immediate (4 hour) verbal	Follow-up to DOE-HQ information (Hazardous Material Release)

TYPE NO. 1 EVENTS

- A. Fatality or imminent fatality due to DOE operation.
- B. Any one occurrence in a DOE operation, that results in 5 or more lost workday cases.
- C. Loss or damage to property over \$100,000.
- D. Occurrence involving a nuclear explosive resulting in an explosion, fire, spread of radioactive material, personal injury or death, or damage to private property.
- E. Accidental release of pollutants which did or could result in significant effect on the public or offsite environment.
- F. Accidental release of hazardous pollutants.
- G. Apparent loss or theft of material (radioactive or nonradioactive) which could be hazardous to individual health and safety.
- H. A single or annual accumulated whole-body exposure of an individual to 25 rem or more of radiation, a single exposure of the skin of the whole body of an individual to 75 rem or more, a single exposure of the forearms of an individual to 150 rem or more.
- I. Internal uptake of radioactive material which, based on early assay data, could result in a dose or dose commitment in excess of five times the pertinent annual standards (25 rem for whole-body dose).
- J. Nuclear criticality in a system not intended to reach criticality.
- K. Release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 5,000 times the respective concentration guides in DOE Order 5480.1, Chapter 11.
- L. Release of radioactive material offsite that could result in an annual dose or dose commitment greater than the annual standards in DOE Order 5480.1, Chapter 11.
- M. Occurrence which may give rise to an inquiry by the public or press of sufficient importance to notify DOE Headquarters.
- N. Occurrence where a press release is made or information is provided to news media, other Federal agencies, or state or local authority.
- O. Radiological assistance occurrence.
- P. Request to an offsite authority for emergency assistance involving a nuclear operation.

- Q. Discovery of significant contamination (radioactive or nonradioactive) onsite or offsite due to DOE operations.
- R. Aviation-related occurrence with fatality, lost workday injuries, explosion or fire, or substantial property damage.

NOTE: Refer to DOE 5484.1 Part 1.1 for more detailed examples.

TYPE NO. 2 EVENTS

- A. Loss or damage to property amounting to between \$50,000 and \$100,000.
- B. Any occupational illness which results in inpatient hospitalization.
- C. Any series of occupational illnesses with the same or similar causes involving five or more persons of which at least one is a lost workday case.
- D. Any occupational illness resulting in more than 5 days away from work.
- E. Any radiation exposure to an individual which in one calendar quarter exceeds any of the following:
 - 1. 5 rem to the whole body
 - 2. 15 rem to skin of whole body or thyroid
 - 3. 30 rem to the forearms
 - 4. 75 rem to the hands or feet
- F. Internal uptake of radioactive material which, based on early assay data, could result in a dose or dose commitment in excess of pertinent annual standards (5 rem for whole-body dose).
- G. Any unplanned nuclear excursion in a reactor, whether or not terminated by protective action, which is clearly outside the routinely accepted and experienced bounds for that reactor.
- H. Any vehicle transporting radioactive material known or found to be contaminated above specified limits.
- I. Any shipment of radioactive material that arrives damaged to the extent that there is substantial reduction in package effectiveness, possible leakage of radioactive contents, or contamination in excess of specified limits.

NOTE: Refer to DOE 5484.1 Part 1.2 for more detailed examples.

TYPE NO. 3 EVENTS

- A. Are internal or external threats to safety, environmental, or health protection or the ability of a facility to operate;
- B. Involve degradation of environmental, safety or health conditions;
- C. Result in serious personnel injury or significant lost workday cases; personnel contamination, assimilation, exposure or significant onsite or offsite contamination of hazardous or radioactive materials in excess of administrative limits but within regulatory limits; or degradation of environmental monitoring equipment necessary to demonstrate compliance;
- D. Result in the violation of safety, environmental, or health administrative limits; or
- E. Involve operational procedural violations, including maintenance and administrative procedures which have the potential to impact the safety, security, environmental or health performance or operation of a facility.

NOTE: Refer to DOE 5000.3A, Attachment 1, for more detailed examples.

TYPE NO. 4 EVENTS

Any significant non-routine release of any pollutant or hazardous substance, e.g., releases of hazardous substances that are reported to the EPA National Response Center as required by CERCLA.

10.11 Control of Equipment and Systems

10.11.1 Background

It is imperative that equipment and systems in a nuclear facility be properly controlled. The operating shift must be aware of how equipment and systems will function for operational purposes. In addition, the proper component, equipment, and system configuration must be established and maintained to satisfy the design bases and the technical specifications for the facility.

At the time of shutdown, an adequate program for control of equipment and systems was not in place at SRS. This contributed to the Central Control Room staff's inability to maintain system component status, and the lack of control over system status contributed to the large pressure transient (water hammer) event that occurred on January 22, 1989.

10.11.2 Restart Criteria

The criteria for determining the adequacy of the control of SRS equipment and systems for restart are described below.

- (1) A program for system status and equipment control shall be developed by the WSRC, using guidelines from the Institute of Nuclear Power Operation (INPO) as set forth by Chapter 8 of the Institute for Nuclear Power Operations report, INPO 85-017 Guidelines for the Conduct of Operations at Nuclear Power Stations, "Control of Equipment and System Status" (Reference 1), and should contain the following elements:
 - o status change authorization and reporting
 - o equipment and system alignments
 - o Technical Specification compliance
 - o equipment deficiency identification and documentation
 - o maintenance work authorization and documentation
 - o equipment post-maintenance testing and return to service
 - o temporary modification control
 - o equipment locking
 - o lockouts/tagouts/clearances
- (2) Maintaining proper plant configuration shall be the responsibility of the Shift Manager who shall authorize status changes to major plant equipment and systems. Status changes to equipment and systems of lesser importance may be delegated by the Shift Manager, to other cognizant shift positions (i.e., control room supervisor or building supervisor). Any change in system configuration shall be formally

documented (e.g., shift logs, tagouts, Technical Specification action statement). The Shift Manager must be kept advised of changes in status of equipment and systems and is ultimately responsible for proper configuration of the plant during the shift. The control room supervisor and operators must be aware of equipment and system status. The Shift Manager shall ensure that all changes in status are communicated to these persons during shift turnover and briefings.

- (3) Individual components for plant equipment and systems shall be checked for proper alignment prior to placing the equipment or system into operation.

WSRC shall commit in their RSIP program to implement INPO guidance with respect to equipment and system alignments (INPO 85-017, Chapter 8, C.2). The following elements shall be included:

- o Alignment checklist(s) shall be used to establish the initial system baseline configuration upon which operations are based.
 - o Alignment checklists shall be developed for system valve lineups, and electrical lineups to include breakers and switches.
 - o Alignment checklists shall provide (1) component/equipment nomenclature, (2) required position, (3) a location for individual documentation for checking of each component and/or equipment, (4) a location for annotating deviations from the required alignment, and (5) a location for senior supervisory review and approval.
 - o System alignments shall be required when restarting from extended shutdowns and major outages.
 - o Records of equipment and systems alignments shall be retained for reference by the operating shift. Administrative controls that document deviations from reference alignment shall be established for records maintenance.
- (4) Administrative controls shall be established to document entry into and compliance with requirements of Technical Specification action statements and limiting conditions for operation. The Shift Manager shall be cognizant of entry into action statements and shall ensure that appropriate actions are taken to correct or mitigate any adverse consequences that arise from that entry. Logs, status sheets, turnover checklists, or other appropriate documentation shall reflect the entry conditions and actions that are taken in response to technical specification requirements. Appropriate operating personnel shall be apprised of limiting conditions for operation and actions for which they may be responsible. Responsible certified personnel shall periodically review the limiting conditions for operation and action statements in effect to ensure that the required actions are met.

- (5) Equipment deficiencies shall be noted by plant operating personnel and identified in the work control system for correction. Methods that identify deficient equipment to operating personnel shall be established.
- (6) The Shift Manager, or his/her certified designee, shall authorize all maintenance activities on equipment important to safety that affect plant operation or change control room indications or alarms. Authorization shall be in writing on the document controlling the work. Documentation of the status of work in progress shall be available in the control room for review by plant operating personnel.
- (7) Equipment shall be tested following maintenance to demonstrate that it is capable of performing its intended function. The testing shall include performance of all functions that may have been affected by the maintenance. The testing shall also verify that the maintenance performed corrected the original problem and no new problems were introduced by the maintenance. Any testing following maintenance shall be specified on the maintenance work order or accompanying documentation (e.g., maintenance procedure). The Shift Manager shall ensure that testing appropriately proves equipment operability.
- (8) Administrative control systems shall be established for installation of temporary modifications such as electrical jumpers, lifted leads, pulled circuit boards, disabled annunciators/alarms, mechanical jumpers, bypasses, temporary set point changes, installed or removed blank flanges, disabled relief or safety valves; and installed or removed filters or strainers, plugged floor drains, and temporary pipe supports.

These control systems shall provide for safety reviews, installation approval, independent verification of correct installation and removal, documentation of the modification, marking of installed modifications, and periodic audits of outstanding modifications. These controls shall also provide for communicating the installation of temporary modifications to the design authority to allow for technical oversight and an evaluation of the impact on current design activities.

- (9) WSRC shall commit in their RSIP program to implement INPO guidance with respect to Equipment Locking, INPO 85-017, Chapter VIII, C.3. This commitment shall use the Equipment Locking Program as specified in the long term criteria.
- (10) The plant Lockout/Tagout program shall ensure that the control room supervisors can effectively determine the plant configuration and can evaluate the impact of tagging or equipment manipulations on the plant configuration. This program shall provide protection for personnel, plant components, and systems during operation, maintenance, surveillance, and modification activities.

10.11.3 Evaluation

ROSI IV.14, "Control of System Status" (Reference 2), states that the Shift Manager and the components handling senior supervisor are responsible for the status control of all the systems within the scope of the ROSI. When the systems controlled by the components-handling senior supervisor connect to the systems controlled by the Shift Manager, both are required to control the status of the valve, breaker, switch, etc. at the interface. The components handling senior supervisor and the Shift Manager are permitted to delegate control of systems to their designates.

ROSI IV.14 states that, for systems under the control of the Shift Manager, the control room supervisor shall maintain the system's status notebook in the control room. This notebook will be used to formally document changes to the system configuration. ROSI IV.11, "Control of Work" (Reference 3), also requires the control room supervisor to log changes in system status in the system status notebook. The control room supervisor is required to authorize all changes in status of the systems directly affecting reactor operations and the fire detection/suppression systems. These systems are listed in Attachment I to the ROSI.

One of the purposes of ROSI IV.11 is to provide specific instructions to assure that the operations organization controls the operational status of critical reactor related equipment, including systems outside the reactor building. This ROSI addresses reporting work progress for maintenance and modification activities and provides instructions for the test coordinator. The Reactor Operations coordinator is required to provide status of work to the control room supervisor about 1 hour prior to the end of the shift. The test coordinator is to return work packages to the control room supervisor at least 1/2 hour prior to the end of the shift for turnover briefings to the oncoming shift personnel. The system status notebook is to be up to date and included in the shift turnover process as described in ROSI III.3, "Shift Assignment/Turnover" (Reference 4). ROSI III.3 requires the Shift Manager to brief the shift prior to turnover. This will ensure that operations personnel are aware of equipment and system status.

The Conduct of Operations Manual describes the use of checkoff lists for specifying the initial alignment of a system. Section 10 then requires use of these completed checkoff lists to provide a baseline system configuration from which operations, startup, and shutdown can be performed. An independent verification is required to be performed on systems identified in ROSI IV.15, "Independent Verification" (Reference 5), to provide a separate confirmation of component position and system alignment.

The Conduct of Operations Manual states that when an item of equipment or a system that is required to be operable to meet the limiting conditions for operation (LCO) becomes inoperable, the plant enters a degraded condition. It further states that the shift management is responsible for recognizing situations when an event or action may cause the plant to enter a degraded condition. ROSI IV.9, "Recording Failure to Meet Limiting Conditions For Operation" (Reference 6), provides the method for recording the failure to meet an LCO and implementation of the associated Technical Specification

Action statements. It also addresses documenting any change in status affecting the Action statement requirements and the return to compliance with the Technical Specification.

ROSI IV.9 requires that each failure to meet an LCO be recorded on an LCO Status Sheet. All active status sheets are required to be placed in an LCO Status Binder, which is maintained in the control room by the control room supervisor. In addition to signing each LCO Status Sheet, the Shift Manager is required to document each failure to meet an LCO in the Shift Manager's log and inform Operations management. ROSI IV.9 also describes the actions to be taken to update the LCO Status Sheets when compensatory actions are taken or completed, restoration to normal status occurs, and for final disposition. Each CCR supervisor is required to review all active LCO Status Sheets on assuming shift and to document this review on the turnover checksheet in accordance with ROSI III.3. The review is to include verification that all required actions have been, or are being, taken.

RD 3.1 (Reference 7) requires employees and vendors working in the reactor areas to initiate a corrective maintenance work request when an equipment or plant deficiency is observed. A condition tag is required to be completed and attached to aid others in locating the equipment and reducing the possibility of duplicate work requests. The Conduct of Operations Manual provides additional guidance for actions operators are required to take when equipment deficiencies are observed.

RD 3.1 states that the Shift Manager shall authorize work activities that are to be performed on, or impact, equipment under his/her control by signing on the reactor work request form. Once started, work status is required by ROSI IV.11 to be reported to the control room supervisor 1/2 hour before the end of the shift for turnover briefings to the on-coming shift personnel. The work control package is returned to the control room supervisor before the end of the shift for re-authorization on the next shift.

RD 3.1 provides the requirements for performance testing after completed work activities and for the acceptance of the work. The procedure describes the implementation requirements for post-maintenance, acceptance, and functional testing. Guidelines for the performance of testing after the completion of an activity are provided. The scope of the testing is required to be based on the extent of the maintenance, new installation, or modification. Furthermore, a satisfactory test is stated as one that verifies the ability of a particular component or system to perform its intended function. The planner is required to obtain and include performance testing requirements, acceptance criteria, checklists, and data sheets from Engineering in the preparation of the work package. Existing procedures, or applicable sections of existing procedures, may be used to prove system operability and that maintenance or modifications were performed properly. Reactor Operations is required to review the work request for the completed maintenance activity and ensure, among other items, that performance tests have been satisfactorily performed.

The Conduct of Operations Manual (Reference 8) addresses the actions necessary to restore equipment to service after maintenance activities. Reactor Operation's action in support of maintenance testing is specified. Detailed information for returning equipment to service is provided to work control personnel. The procedure requires that tests, inspections, or operational verification ascertain that the following have been completed: (1) repairs, modifications, reconnections, realignments, and special checks for the system or item of equipment to be able to perform its intended function; and (2) the equipment or system is energized, filled, vented, or otherwise conditioned to be in operation or available for manual or automatic function.

RD 3.1 Appendix B, "Temporary Modification Control," provides the administrative controls for temporary modifications in a manner that ensures operator awareness, conformance with design intent and operability requirements, and preservation of plant and personnel safety. This document defines a temporary modification as "temporary minor alterations made to plant equipment, components, or systems that do not conform with approved drawings or other design documents." Examples of temporary modifications are listed as lifted leads; electrical jumpers; pulled circuit card; disabled annunciator alarms; mechanical jumpers/bypasses; temporary setpoint changes; installed or removed blank flanges; disabled reliefs, safety valves, or vacuum breakers; installed or removed filters or strainers; plugged floor drains; and temporary pipe supports. The procedure provides detailed guidance regarding limits and precautions, initiation, evaluation, approval, installation, removal, and closeout of temporary modifications. Jumper cables are required to be of "contrasting color to stand out from other cable," and mechanical jumpers "will be uniquely marked to stand out from other hoses or pipes."

A safety review of the temporary modification is required to be performed by Reactor Engineering prior to approval. This review is to ensure that the temporary modification does not adversely affect the intended safety, reliability, or design of the plant. Documentation of the temporary modification is through drawing changes, special instructions, and temporary procedure changes that have been compiled and distributed with the temporary modification package. An independent verification of correct installation and removal is required. Reviews of the temporary modification log each shift and field verifications each month are required by RD 3.1, Appendix B.

The control room supervisors are to determine the plant configuration and evaluate the impact of tagging or equipment manipulations on plant configuration through controls established by the tagout/lockout program, RD 3.1, Appendix A, "Lock, Try, and Tag." A detailed review of the tagout/lockout program is addressed separately in SER Section 10.9.

WSRC and DOE evaluations of the control of the system status program for chargeback determined that it was fragmented and deficient in allowing the operators to maintain adequate status control. Consequently, WSRC revised the requirements and consolidated them into a single document to facilitate control during and immediately following chargeback. As a result, K-Area Standing Instruction D-56, "System Status and Configuration Control," was produced. This procedure contains directions for maintaining control of system status by providing specific instructions to determine if a system,

component, or device is operable or functional. The instruction contains sections on terms and definitions, status assessment, response to inoperable or non-functional status, restoration to operable or functional status, and personnel responsibilities.

The attachments to D-56 provide the means for documenting equipment status and changes to that status. Attachments include: "System Status Checklist"; "Off-Normal Data Sheet"; "List of Systems"; "Audit Sheet"; and "Controlling Procedures." Attachment C, "List of Systems," contains the components and systems that have been determined to be necessary to control in support of chargeback. Additionally, all related tests, inspections, and periodic surveillances required to establish and maintain each component and system operable and/or functional are listed. Using this information and the administrative guidance of the system status checklists, the status of each required system is established. Any changes to items covered by these documents are required to be evaluated for status impact by the Shift Manager and supporting technical personnel, as required.

Upon completion of chargeback, WSRC began revising the requirements contained in K-Area Standing Instruction, D-56, to develop a more comprehensive system status control program for restart. Once the final restart program is developed, WSRC plans to issue it as part of the Conduct of Operations Manual. Pending completion of this new program and review by the DOE staff, this issue will be carried as an open item.

10.11.4 Conclusion

Initial review of the equipment and systems status control program indicated that the current program satisfactorily addresses the restart criteria. However, evaluation of program implementation revealed program weaknesses that prevent effective implementation. As a result, WSRC is revising the program to correct these weaknesses while still maintaining the specific requirements that satisfy the restart criteria. The results of this review and the resolution of the open item will be reported in a supplement to the SER.

10.11.5 References

1. Institute of Nuclear Power Operation, INPO 85-017, Guidelines for the Conduct of Operations at Nuclear Power Stations, Chapter 8, "Control of Equipment and System Status," Rev.1, April 1988.
2. WSRC, DPSOP 336, Section IV.14, "Control of Systems Status," Rev. 0, December 1989.
3. WSRC, DPSOP 336, Section IV.11, "Control of Work," Rev. 3, June 1990.
4. WSRC, DPSOP 336, Section III.3, "Shift/Assignment Turnover," Rev. 7, October 1990.
5. WSRC, DPSOP 336, Section IV.15, "Independent Verification," Rev. 2, September 1990.

6. WSRC, DPSOP 336, Section IV.9, "Recording Failure to Meet Limiting Conditions for Operations", Rev. 0, October 1990.
7. WSRC, RD 3.1, "RRD Work Control Manual," Procedure DM 3.1, Rev. 1, October 1990.
8. WSRC, R 2.1, "Conduct of Operations Manual," Rev. 0, May 1990.

10.12 Independent Verification

10.12.1 Background

Operations consists of several activities, the cumulative effect of which is to ensure safe reactor operations. One of those activities is the development of an independent verification program (IVP) to verify that plant systems and equipment critical to safe reactor operation are configured as intended. Independent verification is the act of checking a component position independent of activities related to establishing the component's position. Not all components require independent verification, because the possibility of mispositioning may be quite remote or because the effect of a mispositioning may not be significant to safe and reliable plant operation. Independent verification recognizes the human element of component operation; that is, any operator, no matter how proficient, can make a mistake. The independent verification process is one that must fully integrate all aspects of plant operations and maintenance personnel involved in the day-to-day operation of plant systems.

At the time of shutdown, no formal independent verification program was in place at SRS. This contributed to the Central Control Room staff's inability to maintain control of system and component status. This was the case in the large pressure transient (water hammer) event that took place on January 22, 1989. Had an independent verification program been in place, the pinned check valve that caused the water hammer event might have been discovered and corrected.

WSRC committed to conform to INPO guidelines associated with the process of independent verification. This includes: clear definitions of terms, clear assignment of duties and responsibilities, instructions on what constitutes independent verification and when it should be used, listing of components and systems requiring independent verification, and training on independent verification requirements and techniques.

10.12.2 Restart Criteria

The criteria for determining the adequacy of the SRS independent verification program for restart are described below.

(1) General

WSRC shall develop an IV Program based on INPO 85-017, Chapter 10, "Independent Verification" (Reference 1), which contains the following elements:

- o clear definition of terms
- o clear assignment of duties and responsibilities
- o instructions on what constitutes independent verification and when it should be used

- o instructions on independent verification techniques
- o listing of systems, equipment, and components requiring independent verification
- o independent verification requirements and techniques training

The independent verification program shall include any group, organization, or individual involved in generating work/job plans for use inside the reactor plant area.

(2) Duties and Responsibilities

The individual/organizational functions of the IV Program shall be clearly defined. For example, the Plant Manager shall:

- o be responsible for the overall direction of the IV Program
- o ensure that operators are trained in IV Program requirements
- o ensure that only qualified personnel perform independent verifications
- o determine and track to completion those corrective actions that need to be taken to resolve discrepancies/problems
- o ensure independent verification documentation is maintained

In addition, the Plant Manager shall be permitted to authorize deviations from normal verification practices should plant/personnel safety dictate.

(3) Systems, Equipment, and Components Requiring Independent Verification

- (a) All systems, equipment, and components (valves, breakers, flanges, etc.) found in Restart Criteria Section 5.6.2.3, Table 5.6-1, Safety-related Equipment List (Baseline Systems), require independent verification. Other nonsafety-related components whose mispositioning or incorrect condition could prevent safety systems from functioning may also require independent verification. The Plant Manager may increase the scope of the independent verification to include non-safety-related components.
- (b) Independent verification should be identified in procedures or checklists by providing dual signoffs. The subject procedure or checklist steps may designate that two individuals sign for an action or may require separate steps for the independent verifications. Where procedures have not yet been revised to reflect the two signoffs, administrative compensatory measures shall be in place to ensure independent verification.

- (c) The WSRC Procedure Writer's Guide shall state the independent verification procedure requirements.

(4) Independent Verification Requirements

Criteria to identify components requiring independent verification are as follows:

- (a) Independent verification of safety-related components and nonsafety-related components that can impact the operation of safety systems shall be evaluated in accordance with the following guidelines:

- o Safety-Related Systems -- All components in systems that have safety-related functions shall be considered for independent verification. However, independent verification of safety-related systems will not be necessary if:
 - mispositioning would not affect system performance
 - mispositioning would be known immediately to an operator
 - significant radiation exposure would be received by the person performing the independent verification and if there is some other compensatory means of ensuring the correct position of the component.
- o Non Safety-Related Systems -- Independent verification shall be performed on those non safety-related systems identified by the Plant Manager as having the potential to:
 - challenge safety-related systems
 - inadvertently cause radioactive releases
 - result in a reactor scram

These systems are usually considered systems important to safety in the commercial nuclear industry.

(b) Occasions Requiring Independent Verification

Independent verification shall be provided for components when the equipment they serve must be available, and a reasonable possibility exists that the components might be mispositioned. The following guidelines describe situations where independent verifications would be appropriate:

- o In returning systems to service following modifications, maintenance, or testing operations, it is essential that the position of components that had been tagged-out to establish

an isolation boundary for the work be independently verified. In addition, other components that may have been mispositioned during maintenance or testing should be selected on a case-by-case basis by the operations individual authorizing equipment return to service. These components are to include those that had been worked on during the maintenance operations. During surveillance tests, components to be tested are normally identified, and the likelihood of their being mispositioned would be remote. Therefore, independent verification would only be necessary for components that are positioned during surveillance tests.

- o To ensure that only the specific items of equipment intended to be removed from service are affected by tag-outs, independent verifications shall be performed when equipment is removed from service. An acceptable verification process in such instances would be to verify that all of the tags were placed on the correct components.
- o During extended outages, it is often impractical to completely restore systems to their normal operating alignment immediately after maintenance is completed. Therefore, restoration of systems not needed for shutdown operations can be delayed until shortly before startup, when all systems whose status is questionable are normally checked.
- o The positions of accessible critical reactor components shall be verified periodically while the plant is operating to prove that the associated equipment is fully operable. Since this check is independent from any activity that might have caused any mispositioning, it can be considered an independent verification, and therefore, a second check would normally not be necessary. If a mispositioning was discovered while the periodic check was being performed, the component position would then be corrected and an independent verification of the corrective action would be required.

(5) Qualifications

The individual(s) assigned to perform component manipulations, system, and valve lineups shall:

- (a) Have completed the training requirements specified by their respective organizational managers, including training on the need for independent verification and on the concepts addressed by the program, as well as on the independent verification procedures and techniques applicable to each particular activity.

- (b) Be qualified by the appropriate organizational manager to perform the test, alignment, or install the temporary modification.
- (c) For operations system lineups, the individual assigned shall be a qualified reactor operator familiar with that system and qualified in the areas of clearance tagging, valve operation, and circuit breaker operation.

(6) Verification Techniques

WSRC shall institute reactor operations staff training in the techniques for verifying the positions of all plant components. Reference instructions shall additionally be provided explaining what an operator shall do when verifying a component position. WSRC's Operations organization shall use its most experienced personnel to develop these instructions using applicable manufacturer's recommendations where available. The instructions shall describe the techniques for verifying manual valves, motor-and air-operated valves, circuit breakers, blank flanges, removable links in fuses, solenoid-operated valves, and the availability of control power. The instructions should be designed to ensure that operators will use approved methods when verifying component positions and be a tool for ensuring uniformity in operator performance and performance measures. The following general guidelines should be considered in the development of independent verification training modules and implementing instructions:

- (a) Independent verifications shall be conducted in a manner such that each check constitutes an actual identification of the component and a determination of both its actual and required positions. To be independent, the integrity of the checks must be maintained by minimizing interaction between the personnel who operate the components and the personnel who perform the independent verifications. Physical separation of the individuals need not be required, but this could avoid non-independent verification pitfalls. An example of such a pitfall would be a verifier observing an individual operate a valve while relying on the operator to correctly read the valve label or procedural requirement. It is not always possible to determine if an operator has completely shut a valve by merely observing the action, and mistakes in component identification or requirement determination may not be caught without both individuals reading the labels and procedures.
- (b) Equipment failures have occurred that caused remote valve positioning indicating lights on control panels to be incorrect. For this reason, it is desirable that one of the checks be made locally at the component or from the main control panel indicating lights. It is unlikely that a single failure would cause both remote indicators to be in error. Therefore, the use of different remote indicators for the same component would improve the reliability of the independent verification. Additionally, the human factors associated with verifying component positions also

make verifications using the same indicator or method highly subject to common-mode failures.

- (c) In some situations, a component's position can be determined by observing process parameters such as pressure, flow, or voltage. The observation of process parameters, combined with a physical check of a component's position, can constitute an independent verification. However, care must be exercised when using process parameters, because alternate flow paths or other factors could cause them to be misleading indicators of component position. For example, voltage on a bus would prove that a particular supply breaker was shut only when no alternate power supply exists. Additionally, flow and pressure do not necessarily indicate that a valve is fully open. For these reasons, process parameters alone should not be relied upon as an indicator of a component's position. The plant operating procedures shall specify where and when process parameters are acceptable indicators of component position.
- (d) Verifying throttled valves by shutting and reopening the valve a prescribed number of turns shall be avoided because this practice has the potential to create valve mispositionings. Instead, position indicators, scribe marks, or other recognized and designated indicators shall be used to determine throttled valve positions. When shutting and reopening a throttled valve is necessary to determine its position, having another person observe the valve operator's actions would be preferred to having both persons independently shut and reopen the valve.
- (e) In some cases, surveillance testing may be substituted for normal verification techniques in checking that components are correctly positioned. A notable example would be a full-flow test to prove the correct positioning of flow control valves. However, surveillance tests frequently will not serve to verify the position of all components that are important to subsequent system operation. For example, running a pump in recirculation would not prove that all main flow path valves were properly positioned, or that support functions are aligned properly. Therefore, surveillance testing shall not be used as independent verification unless the method prescribed in item (a) above is impractical and it can be shown that the test conclusively proves the position of the components in question.

10.12.3 Evaluation

ROSI IV.15, "Independent Verification," (Reference 2) establishes guidelines for the implementation of an independent verification program within the reactor operations organization. The ROSI defines independent verification as examinations or measurements performed by qualified individuals who did not perform the initial operation or activity. Independent verifications are

normally separated from the initial operation by time and distance, and exceptions to the separation by time and distance requirement require approval by the Shift Manager.

ROSI IV.15 states that the Manager, Reactor Operations, is responsible for (1) establishing the IVP for reactor operations personnel, (2) establishing the training requirements necessary to qualify personnel for conducting independent verifications, (3) ensuring all managers in operations are aware of and familiar with the IVP, (4) providing liaison with other organizations to ensure uniformity of requirements, and (5) designating the systems and/or components that shall require independent verifications. The Plant Manager is responsible for ensuring that independent verification requirements are strictly adhered to within each reactor plant area. The Work Control Supervisor is responsible for reviewing all work packages to ensure that the requirements of this procedure have been addressed. This ensures that any group, organization, or individual involved in generating work plans for use inside the reactor plant areas is included in the IVP. Additional responsibilities are listed for the Operations Manager, Shift Manager, Control Room Supervisors/Operators, Reactor Building Supervisors/Operators, and the Reactor Training and Procedures group.

ROSI IV.15 includes the qualifications required for individuals assigned to perform independent verifications. These individuals are required to (1) have completed the training requirements established by the Manager, Reactor Operations, including IV procedures and techniques; (2) be on the Approved IV List; (3) for operations system lineups, be a Certified Reactor Supervisor, Certified Central Control Room Operator, Qualified Reactor Building Supervisor, or Qualified Reactor Building Operator; and (4) for tagouts, have received training on DMP-3.01, Appendix A, "Lockout" (Reference 3).

ROSI IV.15 states that all breakers, valves, and components of safety-related systems listed in DPSOL 105-1829A-K, "Significant Equipment Lists" (Reference 4) are subject to independent verification, as are nonsafety-related systems that:

- o have the potential to challenge safety related systems
- o have the potential to inadvertently cause a radioactivity release
- o have the potential to result in a spill or release of hazardous material to the environment
- o involve the alignment of electrical breakers, switches, fuses, etc.
- o are designated by the Plant Manager and/or approved procedures

The staff determined that the listing of systems does not fully meet the restart criteria in that ROSI IV.15 only requires independent verification on systems listed in DPSOL 105-1829A-K, which lists only 7 of the 13 systems contained in SER Section 5.6.2.3, Table 5.6-1, Safety-Related Equipment List (Baseline Systems). Pending either a revision to the ROSI to require

independent verification on the required 13 systems or a revision to DPSOL 105-1829A-K to include all 13 systems, this issue remains an open item.

ROSI IV.15 specifies that all breakers, valves, and components that are subject to the requirements of independent verification shall be in the correct position/condition for the following situations:

- o during initial system lineups following an outage where the system was not maintained in a normal operating lineup
- o when a system or component is being returned to service or restored to a standby lineup and all tags are being removed
- o when directed by specific surveillances or procedures
- o anytime valves or components of the Process Water System are manipulated
- o anytime a valve or component identified by the plant manager as requiring independent verification is restored to a normal configuration

In addition, ROSI IV.15 specifies that the following valves and components shall be subject to independent verification:

- o all valves, breakers, and other components in safety-related systems in which an incorrect positioning could affect the system operation adversely, or result in a SCRAM
- o all valves, breakers, and other components that, if incorrectly positioned, could result in the release of radioactive material to the environment
- o components with a reasonable possibility of being mispositioned (i.e. may have been manipulated since last alignment)

ROSI IV.15 specifies that all tagouts involving safety-related systems; nonsafety-related systems with the potential to challenge safety-related systems, inadvertently cause radioactive releases, result in a reactor SCRAM, or cause a spill or release of hazardous material to the environment; or the alignment of electrical breakers, switches, fuses, etc. shall have an independent verification performed both for the installation of the tagout and the removal for return to service.

The staff determined that the occasions requiring independent verification do not fully meet the restart criteria. ROSI IV.15 does not address all occasions for which independent verification is required. Specifically, ROSI IV.15 does not address periodically verifying the positions of accessible critical reactor components while the plant is operating to prove that the associated equipment is fully operable. Pending revision of ROSI IV.15 to address all occasions requiring an independent verification, this issue remains an open item.

ROSI IV.15 provides independent verification techniques for unlocked valves and circuit breakers. Unlocked valves are required to be checked in the "close" direction only. The ROSI allows the IV operator to use scribe marks or other position indicators if available or to observe the initial positioning of a throttled valve, rather than manipulating the valve a second time. The procedure also states that use of process parameters to verify component position is not the preferred methodology and should be used only if direct means of verification are not available.

The ROSI does not include guidance for performing independent verifications on components included in lockouts. Instead, the ROSI defers to the lockout procedure, RD 3.1, Appendix A. This procedure discusses the performance of independent verification in the process of establishing the lockout. Refer to section 10.9 for more information on this subject.

WSRC's "Conduct of Operations Manual," R2.1 (Reference 5), states that throttled valves may be verified visually to be in position. ROSI IV.15 provides alternate verification techniques that may be used at the discretion of shift supervision. These include the use of remote indicators, process parameters (such as flow, pressure, and voltage), functional mechanical indicators, and valid functional testing (such as with a voltmeter). The Conduct of Operations manual states that care must be exercised when using process parameters as a second check of a component's position due to possible alternate flow paths or other conditions that could make this method unreliable.

ROSI IV.15 provides general guidance for the implementation of the independent verification program. Included in this guidance are the requirements that the act of independent verification be completely separate from the initial activity and that personnel performing the verifications minimize interaction with each other so as not to bias the process. The ROSI states that independent verification shall ensure that each check constitutes an actual component identification and a determination of required and actual position/condition.

The Conduct of Operations manual states that position indicators are subject to equipment failures that could result in displaying the incorrect status of a valve or breaker. Personnel are directed to perform one check locally at the component to avoid positioning errors due to indicating equipment failures. ROSI IV.15 grants the shift manager the authority to modify independent verification requirements if excessive radiation exposure or other personnel hazard would result. This would be accomplished using a temporary procedure change or other approved processes.

DPSP 85-1090, "Procedure Writer's Guide" (Reference 6), addresses the inclusion of signatures for procedures dealing with systems included in the IVP. If it is determined that a procedure or job plan needs independent verification, but has not been revised to include the proper sign-offs, ROSI II.8, "Procedure Compliance" (Reference 7), provides the steps required to document the independent verification.

10.12.4 Conclusion

The staff has identified open items in the IVP as discussed above. Upon satisfactory resolution of these open items, the staff will be able to conclude that the IVP is acceptable for restart. The resolution of those open items will be addressed in a supplement to the SER.

10.12.5 References

1. Institute of Nuclear Power Operations, INPO 85-017, Guidelines for the Conduct of Operations at Nuclear Power Stations, Chapter 10, "Independent Verification," Rev. 01.
2. WSRC, DPSOP-336, Reactor Operations Standing Instruction IV.15, "Independent Verification," Rev. 2.
3. WSRC, RD 3.1/DMP 3.01, "Work Control." Rev. 0.
4. WSRC, DPSOL 105-1829A-K, "Significant Safety Equipment Lists," Rev. 0.
5. WSRC Reactor Operations, "Conduct of Operations Manual," Rev. 0.
6. WSRC, DPSP 85-1090, "Reactor Training and Procedure Writer's Guide," Rev. 9.
7. WSRC, DPSOP-336, Reactor Operations Standing Instruction II.8, "Procedure Compliance - Reactor Department Procedures," Rev. 6.

10.13 Fitness for Duty

10.13.1 Background

A Fitness-For-Duty program is an essential element of facility safety. In addition to protecting facility property and providing for public safety, this program ensures that facility personnel are provided with a safe work environment. Substance abuse of any kind is considered a direct threat to employee well-being and to the safety of the community at large. Agencies of the Federal Government have expanded the acceptable standards of fitness-for-duty to include the prohibition of illegal drugs from the work place.

There was no formal Fitness-For-Duty program for SRS in the past, although comprehensive physical health and drug/alcohol screening programs have been used for several years.

10.13.2 Restart Criteria

A schedule for the implementation of a Fitness-For-Duty Program that meets DOE orders shall be submitted to the staff for approval before restart. The criteria for determining the adequacy of the SRS Fitness-For-Duty Program for restart are described below.

(1) Policy

- (a) A clear, written policy on drug use, alcohol abuse, and the physical and psychological condition of workers shall be written and issued prior to restart. The Fitness-For-Duty Policy is to be applicable to all WSRC employees who direct or operate reactor systems or reactor components. Prior to being granted unescorted access to the reactor facility, chemical testing shall be required for all WSRC employees, visitors, contractors, or vendors performing services at SRS.
- (b) The Fitness-For-Duty Policy shall emphasize the importance of employee health and safety as related to safe plant operation. Stated policy shall address the following:

Specify that co-worker involvement is needed in identifying and reporting a worker suspected of not being fit for duty.

Specify that the use, sale, or possession of drugs while onsite shall be clearly prohibited.

Specify that personnel who use, sell, or distribute drugs onsite are declared ineligible for further unescorted access to the reactor facility.

Specify that personnel who have been convicted of selling or distributing drugs offsite or who have a history of repeated drug use offsite shall be prohibited from unescorted access to the reactor facility.

Specify that, when personnel are suspected of using drugs, chemical testing (for-cause and random tests) for drug involvement is required before granting access to the reactor facility. No more than 2 years shall elapse between chemical testing, including random testing, testing for cause, or testing as part of a routine physical examination.

(2) Senior Management Support

- (a) The Office of the President, WSRC, shall endorse and issue an effective Fitness-For-Duty policy that demonstrates management support.
- (b) Management support shall be evidenced by the development of a clear Fitness-For-Duty policy and by its application to specified WSRC employees, visitors, and other contractor or vendor personnel performing services for WSRC.

(3) Training

- (a) WSRC supervisory personnel shall be trained in the overall policy and specific requirements of the Fitness-For-Duty policy.

Training shall be conducted when the program is first implemented or when major program changes are made, Training shall also be conducted on a recurring basis for refresher purposes.

New employees or individuals who receive unescorted access to the reactor facility (such as visitors or contractors) shall acknowledge their understanding of the requirements of the Fitness-For-Duty policy.

- (b) WSRC supervisory personnel shall be provided with appropriate health and safety information on drugs and alcohol, which shall include the following:
 - understanding chemical test results and the effects of prescription and over-the-counter medications, as well as dietary conditions, on job performance.
 - techniques for recognizing drugs and indications of the use, sale, or possession of drugs.
 - techniques for recognizing unusual behavior.
 - procedures for reporting problems to supervisory or security personnel.

- details regarding the Employee Assistance Program available to WSRC employees.
- the role of the site medical staff.

(4) Policy Implementation

- (a) Procedures should provide guidance to managers, supervisors, and other key personnel on the necessary actions to take if an employee appears to be unfit for duty or is suspected of possessing or having used drugs or alcohol while onsite. The procedure shall include instructions on the following:
 - how to recognize aberrant behavior.
 - how to confront/intervene when an employee is suspected to be unfit for duty.
 - how to remove personnel suspected to be unfit for duty from protected areas.
 - how to initiate medical observation and chemical testing.
 - requirements to be implemented when sending an employee home.
- (b) Supervisors shall be trained to take initial actions correctly and promptly without guidance from senior management (who may not be available). Training shall:
 - emphasize the importance of timeliness when addressing an aberrant behavior situation.
 - focus on actions to be taken by supervisors while awaiting direct assistance from management and other appropriate personnel.
- (c) Guidelines, such as those listed below, shall be in place regarding how to handle personnel whose offsite drug use is pending resolution of the case.
 - prompt removal from duty of personnel with confirmed evidence of drug use, where resolution of the case is pending.
 - assessment of the extent of the person's drug involvement and potential for rehabilitation.
 - senior management review and decision (based on professional assurance that rehabilitation efforts are successful) regarding suitability for the person's return to duty.

- appropriate probationary period for personnel who have returned to duty under close supervisory observation and chemical testing to ensure continued freedom from drug use.

- (d) A monitoring program that includes chemical testing and observation of employees is significant in the achievement of a drug-free work environment.

(5) Chemical Testing

- (a) All WSRC employees, visitors, contractors, or vendors performing services for WSRC at SRS shall have successfully completed chemical testing prior to being granted unescorted access to the reactor facility.
- (b) Chemical testing (for-cause tests) shall be initiated when management has cause to suspect degraded performance, unusual behavior, or a work place incident that may be related to substance use.
- (c) Chemical testing of personnel shall be conducted at the commencement of employment and on a random basis in accordance with a WSRC-approved random sampling program.
- (d) The site medical facility shall have appropriate procedures for obtaining bodily fluids for testing, accountability of specimens, and laboratory methodology, including analysis of sensitivity levels, and quality control.
- (e) The test methods used in the laboratory should reflect recognized industry standards and shall ensure the highest degree of reliability in the test results.
- (f) Periodic evaluations of laboratory performance shall be conducted by WSRC.

(6) Employee Assistance Program

- (a) The availability of assistance and rehabilitation for personnel involved with chemical dependency can strengthen a Fitness-For-Duty policy. Assistance of this type should contribute toward achieving a drug-free work environment and workforce.
- (b) Employees shall be made fully aware of the availability of any services offered by the program.
- (c) Confidentiality shall be maintained for personnel who use services offered by the Employee Assistance Program. However, a mechanism should be in place to inform appropriate supervisory personnel if a person is determined to be unfit for duty while receiving employee assistance services.

(7) Medical Certification

- (a) Medical certification requirements shall be in accordance with American Nuclear Society (ANS) 3.4, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants," (Reference 1) and Nuclear Regulatory Commission (NRC) Regulatory Guide 1.134, "Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses" (Reference 2). Form NRC 396, "Certificate of Medical Examination," or an alternative form may be used.
- (b) Individuals for whom medical certification is required are those WSRC employees or contractors who direct or operate reactor systems or reactor components (certified supervisors and operators and qualified reactor operators).
- (c) Psychological fitness shall be demonstrated in part by a program of continuous observation by supervision to detect intentional or unintentional acts of aberrant behavior and by a supervisor's evaluation report submitted to the medical examiner prior to a recertification physical. This supervisor's evaluation shall include information specified by the medical staff and shall address areas such as work performance, attendance, behavior changes, incidents of ineptness, poor judgement, lack of physical or emotional stamina, unusual absence or tardiness, and potential psychological medical problems.

(8) Psychological Fitness

Initial screening for psychological fitness is ascertained by granting of the DOE security clearance and a satisfactory pre-employment physical that allows unescorted access.

(9) Specific Reactor Restart Items

- (a) The site-wide Fitness-For-Duty program and implementation schedule shall be submitted to DOE prior to restart.
- (b) The provisions of paragraphs 5a, 5b, and 5c of this section shall be implemented at reactor restart for those WSRC employees, visitors, contractors, or vendors performing services for WSRC who are granted unescorted access to the reactor facility (this is applicable to each reactor facility at restart of each specific facility).
- (c) The provisions of paragraph 7 of this section shall be initiated such that all personnel for whom examinations are required have been examined by April 1, 1992.
- (d) The provisions outlined in paragraph 6 of this section shall be implemented prior to restart.

10.13.3 Evaluation

Reactor Safety Improvement Program (RSIP) A013, Project Control Plan, "Fitness for Duty and Formal Standards for RRD" (Reference 3), has been drafted by WSRC to establish a schedule for implementation of the post-restart Fitness-For-Duty requirements. The project objectives listed are: (1) implement post-reactor- restart Fitness-For-Duty program in accordance with DOE directives and with DOE approval; (2) implement a training program for all employees that will ensure knowledge of reactor Fitness-For-Duty policies and procedures and emphasize the importance of the Fitness-For-Duty program; (3) implement a training program for management/supervisory personnel that will, within practical limitations, allow recognition of employee behavioral aberrations and/or physical signs or changes of an employee that might indicate substance abuse; and (4) develop procedures to ensure that personnel covered by the Fitness-For-Duty program undergo physical examinations every 2 years prior to certification/recertification. Attachment 5 of the Project Control Plan contains the proposed schedule; however, it has not been submitted to the DOE staff for approval. An implementation status document was provided to DOE on October 17, 1990. This document lists the milestone dates for specific SER items, the responsible organizations, and the implementing documents. This document is not, however, an implementation schedule.

The DOE staff has determined that A013 Project Control Plan does not fully meet the restart criteria in that the Fitness-For-Duty implementation schedule has not been approved and submitted to DOE. Pending approval of the schedule by WSRC and subsequent submittal to the DOE staff, this issue remains an open item.

The construction contractor has recently issued a separate substance abuse program, which is being evaluated by the staff.

Pending completion of the evaluation of the Construction Substance Abuse Program to ensure that it meets the restart requirements, this issue will remain an open item.

The policy addressing a Fitness-For-Duty program has been developed by the contractor. The policy is applicable to all employees who direct or operate the reactor systems or components as stated in MP 2.4, "Fitness-For-Duty" (Reference 4).

MP 2.4 has not been approved and issued. Pending approval and issuance of MP 2.4, this issue remains an open item.

The policy requires chemical testing for all WSRC employees, visitors, contractors, or vendors performing services for WSRC at SRS prior to being granted unescorted access to the Reactor facilities. The policy also stresses the importance of employee health and safety. The policy states "WSRC shall seek to provide a work environment free of the negative influences...that jeopardize safe work practices required to...[provide] safe operations...." The policy also states that "The company will support an individual's effort (to rehabilitate) by providing help in identifying the treatment designed to

facilitate rehabilitation." The need for co-worker involvement in identifying personnel suspected of being unfit for duty is included in the policy. The policy encourages personnel to identify and report workers suspected of not being fit for duty. The use, sale, or possession of drugs is prohibited while onsite, and further unescorted access to the reactor facility is denied to personnel or repeated users who use, sell, or distribute drugs onsite or offsite.

MP 2.4 requires employees to undergo drug and alcohol testing on a routine, random, or "for cause" basis to detect and discourage the use of illegal drugs and/or alcohol. WSRC-1-03, "Management Requirements and Procedures-Fitness for Duty" (Reference 5), states that "The period between chemical tests whether from routine physicals, for-cause, or random, shall not exceed two years for unescorted access to the reactor areas." The policy demonstrates senior management support, as it has been endorsed and issued by the office of the President, WSRC. The Fitness-For-Duty Policy will be approved by the president's office. This policy, as outlined in WSRC-1-03, lists seven specific items for which management is responsible. The policy clearly defines the applicability of the program to visitors, contractors, subcontractors, and vendors performing service for WSRC.

While the restart requirements addressing senior management support of this program and its applicability are met by WSRC-1-03, this procedure has not yet been approved and issued. Pending approval and issuance of this procedure, this issue remains an open item.

The policy dictates training and acknowledgement of understanding of the requirements of the overall policy and specific requirements of the Fitness-for-Duty policy for WSRC supervisory personnel, new employees or personnel receiving unescorted access to the reactor facility. WSRC-1-03 also requires that training be conducted when the program is initially implemented, when major program changes are made, and on a recurring basis for refresher purposes.

The training program ensures that supervisory personnel are provided with health and safety information on drugs and alcohol, including techniques for recognizing drugs and indications of use or possession, techniques for recognizing unusual behavior, procedures for reporting problems to supervisory or security personnel; the Employee Assistance Program available to WSRC employees, and the role of the site medical staff.

The DOE staff has determined that the training program does not meet the restart criteria in that the training program does not address the understanding of chemical test results or the effects of over-the-counter medications and dietary conditions on job performance. There are also no techniques described for recognizing the sale of drugs. In addition, the training program does not provide details of the Employee Assistance Program. Also, the training does not address non-WSRC employees who are required to receive this training. Pending the inclusion of these requirements into the training program, these deficiencies will remain an open item.

In Attachment 1 to WSRC-1-03, the Fitness-for-Duty Policy provides guidance to managers, supervisors, and other key personnel on the necessary actions to take if an employee appears to be unfit for duty or is suspected of possessing or having used drugs or alcohol while onsite. This attachment provides specific characteristics that may indicate aberrant behavior. RD-1, RDP, "Fitness-For-Duty" (Reference 6), includes steps referring to Attachment 1 for recognizing aberrant behavior. Attachment 2 of the RDP addresses confronting/intervening and removing an employee suspected of being unfit for duty. Requirements for reporting problems to management, security or other appropriate personnel, initiating medical referral and chemical testing, and requirements to observe when sending an employee to medical prior to being removed from site are also addressed in Attachment 2. The training program also contains information relating to all of the above items.

RD-1 RDP, "Fitness-For-Duty," meets the restart criteria concerning implementation of the Fitness-For-Duty program. However, RDP has not yet been approved or issued. Pending approval and issuance of this procedure, this issue remains an open item.

Training is provided to supervisors in guidance for the initial prompt and correct actions to be taken without guidance from senior management. WSRC-1-03 provides specific direction for training to be performed and the contents of the training materials including emphasis on the importance of timeliness when addressing an aberrant behavior situation and the actions to be taken by supervisors initially and while awaiting direct assistance from management, security, site medical and/or other appropriate personnel.

WSRC-1-03 provides guidance on the development of guidelines which will address each of the SER requirements involving offsite drug use. WSRC has stated that there are legal ramifications which need to be resolved before these guidelines can be developed.

Pending resolution of these concerns and development of the required guidelines concerning offsite drug use, this issue remains an open item.

The Fitness-for-Duty Policy includes chemical testing and observations of employees. The policy states that "All WSRC employees, visitors, contractors, or vendors performing services for WSRC at SRS shall have successfully completed chemical testing prior to being granted unescorted access to the reactor or site facilities." Westinghouse employees are tested prior to badging, in accordance with WSRC Interoffice Memo ESH-MED-900408, which requires new employees to undergo testing that is documented using a "Substance Abuse Test Verification Required for Badging" form (SAPT). RD-1 RDP states that RRD managers and supervisors will ensure that all WSRC, visitors, contractors, subcontractors, or vendor personnel performing services in the reactor areas have successfully completed chemical testing at the time of initial employment. Additionally, employees will be required to undergo drug and alcohol testing on a routine, random, or "for cause" basis to detect and discourage the use of illegal drugs and/or alcohol. RD-1 RDP provides the detailed steps for the initiation of "for cause" testing. Additionally, Medical Department Manual Q3.6 (Reference 9), Procedure 1007 specifically addresses "for cause" testing.

At the present time, non-WSRC employees are not required to participate in the initial screening or the routing, random, or "for cause" testing program. A method to implement the policy for non-WSRC employees is necessary. Pending development of a method for implementing this policy for non-WSRC employees, this issue is an open item.

WSRC-1-03 calls for the development of procedures for obtaining body fluids for testing; accountability of specimens; and laboratory methodology, including analysis of sensitivity levels and quality control. WSRC Medical Department Manual Q3.6, procedures 1002, 1003 and 1004 satisfactorily describe samples and collection. WSRC-1-03 also requires that test methods be developed that conform to recognized industry standards and ensure the highest degree of reliability in the test results. The test methods and drug levels for screening and confirmations are found in the scope of work proposed by WSRC. The service contract also states the testing methods and drug levels.

The policy described in WSRC-1-03 states that periodic evaluations of laboratory performance shall be conducted by WSRC; however, procedures do not exist that require this evaluation. WSRC currently uses a National Institute on Drug Abuse certified laboratory which, by contract, must notify WSRC if their certification is lost.

The lack of a program to periodically verify that the laboratory remains certified is an open item.

An Employee Assistance Program (EAP) is provided through a private contractor. Provisions to make employees fully aware of the services offered by the program are found in WSRC-1-03. RD-1 RDP requires that the manager/supervisor confronting an individual discuss the potential of referral to the WSRC EAP (for WSRC employees) or the individual's company EAP.

Maintaining confidentiality during assistance is assured through compliance with WSRC-1-03. This procedure requires that confidentiality be maintained for all personnel referred to the medical department or requesting services offered by the Employee Assistance Program. RD-1 RRD specifically states the confidentiality requirements associated with the EAP. WSRC-1-03 also requires a mechanism be in place for keeping supervision informed if an employee is found to be unfit for duty during assistance.

The DOE staff has determined that this issue does not fully meet the restart criteria. Although RD-1 RRD requires a mechanism be put in place to keep supervisors informed if an employee is found unfit for duty during assistance, the program procedures do not specify or describe this mechanism. Pending development of such a mechanism, this issue remains an open item.

A medical certification process is included in WSRC-1-03 which contains requirements that procedures be developed in accordance with American Nuclear Society (ANS) Standard 3.4, "Medical Certification and Monitoring of Personnel Requiring Operator Licences for Nuclear Power Plants," and NRC Regulatory Guide 1.134, "Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses." WSRC Medical Department Manual Q3.1 (Administrative) Procedure 1004 fulfills this requirement. Attachment A, "Certification of

Medical Evaluation," restates the requirement to follow the American National Standards Institute/American Nuclear Society (ANSI/ANS) standard. WSRC-1-03 states that NRC Form 396, "Certificate of Medical Examination" or a suitable alternative may be used. The form actually used is found in the WSRC Medical Department Manual, WSRC Q3.6-Substance Abuse Testing, Procedure 1004.

A psychological certification process is included in WSRC-1-03, which requires that psychological fitness be demonstrated by a program of continuous observation by supervision to detect intentional and unintentional acts of aberrant behavior. RD-1 RDP requires supervisors to document employee work performance and directs RRD managers and supervisors to use Attachment 1 to identify actual or potentially erratic, aberrant, or unsafe behavior. WSRC-1-03 requires the submittal of an evaluation report by supervision to the medical examiner prior to a recertification physical. This report is required by the procedure to include information specified by the medical staff, such as work performance, attendance, behavior changes, incidents of ineptness, poor judgement, lack of physical or emotional stamina, unusual absence or lateness, potential psychological or medical problems. Attachment B of the Medical Department procedure (Q3.1, 1004) also lists each of the above items (less medical department input) as "Facility Operators Report," which is included as part of the medical certification.

WSRC-1-03 states that the "Initial screening for psychological fitness is ascertained by granting of the DOE security clearance and a satisfactory pre-employment or certification physical examination" as required by the SER.

10.13.4 Conclusion

The staff has identified open items in the Fitness-For-Duty Program as discussed above. Upon satisfactory resolution of these open items, the staff will be able to conclude that the Fitness-For-Duty Program is acceptable for restart.

10.13.5 References

1. American Nuclear Society, ANS 3.4, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants."
2. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.134, "Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses."
3. WSRC, RSIP Project A013 Project Control Plan, "Fitness for Duty and Formal Standards for RRD," Rev. 0, May 1990.
4. WSRC, WSRC-1-01, Management Policies, MP 2.4, "Fitness for Duty" (draft).
5. WSRC, WSRC-1-03, Management Requirements and Procedures, "Fitness-For-Duty" (draft).

6. WSRC, RD-1 RDP, "RRD Admin - Fitness-For-Duty" (draft).
7. WSRC, Medical Department Manual Q3.6-Substance Abuse Testing, Procedures 1002, 1003, 1004, and 1007.

10.14 Piping and Instrumentation Diagrams in the Central Control Room

10.14.1 Background

To perform their jobs properly, operations personnel must use up-to-date piping and instrumentation diagrams (P&IDs) that reflect actual system configuration. Without access to controlled as-built plant drawings, operators cannot properly question system indications and responses. Operations personnel must use P&IDs when they prepare tagouts, isolate systems, troubleshoot system problems, and verify valve and instrument lineups. Up-to-date drawings are also required for ensuring proper work control during maintenance and plant modifications.

There were no P&IDs for the SRS reactor facilities at the time of reactor shutdown; instead, operators used procedure schematics when performing their work assignments. These schematics included the equipment referenced in the associated procedures, but did not identify all components within a system; therefore, it was difficult, if not impossible, to obtain accurate knowledge of system status. This inadequate knowledge of system status contributed to the cooling water pressure transient (water hammer) which occurred in K-Reactor on January 22, 1989.

WSRC committed to issue system drawings (both P&IDs and electrical one-line diagrams) to the control room based on a verified system configuration and identified 13 safety system P&IDs that will be established in the control room prior to restart.

10.14.2 Restart Criteria

The criteria for determining the adequacy of control room information for restart are described below.

- (1) A controlled drawing file containing as-configured P&IDs and electrical one-line diagrams is to be established and placed in the control room. The control room drawing file is to contain drawings for the following systems: (1) Scram, (2) Cooling Water, (3) Emergency Electrical Power, (4) 125-Vdc Power, (5) Process Water (PW), (6) Control Rod and Rod Reversal, (7) Central Control Room (CCR) Habitability, (8) Supplementary Safety, (9) Water Removal and Storage, (10) Emergency Cooling, (11) Process Room Spray, (12) Discharge Assembly Cooling, and (13) Process Water Diesel Generators.
- (2) Procedures shall be developed and implemented to ensure that all modifications that result in a physical change to the plant operational configuration are incorporated into the controlled drawings. (The evaluation of the procedures for ensuring that the subject drawings are properly controlled and updated is discussed in Section 5.5 of this report).
- (3) Operations personnel shall be trained on how to read, understand, and use the subject P&IDs and one-line diagrams.

10.14.3 Evaluation

ROSI II.13 (Reference 1) provides detailed instructions for the control of drawings in the control room. The procedure requires that all drawings used must be from the controlled file or be verified as the latest revision through the document control center. The controlled file will automatically receive updated drawings from the document control center. An examination of the central control room (CCR) drawing file found that the following drawings were present and marked as controlled:

S5-1-9844 through 9847	P&ID Symbols and Legends
S5-1-9848	WRS Sump Pump Bearing Cooling Wtr
S5-1-9849	WRS
S5-1-9850	PWDG Cooling Water Sys Bldg 108-1
S5-1-9851	PWDG Cooling Water Sys Bldg 108-2
S5-1-9852	PWDG Fuel Oil Sys Bldg 108-1
S5-1-9853	PWDG Fuel Oil Sys Bldg 108-2
S5-1-9854	PWDG Lube Oil Sys Bldg 108-1
S5-1-9855	PWDG Lube Oil Sys Bldg 108-2
S5-1-9856	Discharge Assembly Cooling
S5-1-9857	Process Room Spray
S5-1-9858	ECS
S5-1-9859	ECS
S5-1-9879	Scram Logic
S5-1-9880	Scram Simplified Circuit Diag
S5-1-9881	Scram Simplified Channel Circuit Diag
S5-1-9875	Central Control Room Habitability HVAC
S5-1-9875	Central Control Room Habitability Lighting
S5-1-9875	Central Control Room Habitability HVAC Pipe
S5-1-9860 through 9864	Cooling Water
S5-1-9877 through 9878	Cooling Water
S5-1-9882	Control Rods Logic Diagram
S5-1-9883	Control Rods Simplified Circuit Diag
W2008592	125 Safety Related DC Bldgs 151-1,2, 190
SD-5-1-2005602	125 Safety Related DC Single Line
S5-1-9865 through 9868	Process Water
S5-1-9873 through 9874	Supplementary Safety
S5-1-9870	GM Diesel Fuel Oil Supply
S5-1-9871	GM Diesel Lube Oil Supply
S5-1-9872	GM Diesel Cooling Water
S5-1-9876	Recirculation Pump Diesel
SD-5-1-2005598	Transformer Room
SD-5-1-2005599	Transformer Room
SD-5-1-2006958	
through 2006968	MCC's
SD-5-1-2006917	Recirculation Pump Diesel S/L

Sections 3.9.6, 5.2.4., and 6.6, (Reference 6) and ASME Code Section XI to the extent practicable. Any exceptions of required examinations and testings from ASME Section XI shall be identified, evaluated, and approved by DOE. A description of the programs and the implementation schedule shall be submitted to DOE for review and approval prior to restart.

14.3 Evaluation

The staff reviewed applicable documents and conducted program reviews to determine if WSRC's programs or plans meet the inspection and testing restart criteria. The staff evaluated a number of reports and programs related to inspection and testing programs, quality assurance programs, and the reactor tank ultrasonic test program (Reference 7). The staff was assisted in this review by a panel (DOE Independent Review Committee [DIRC] for System Integrity) of nationally recognized experts in the areas of metallurgy, fracture mechanics, corrosion, and nondestructive examination (NDE). The results of the staff's review are discussed below.

14.3.1 Scope and Issue Definition

Restart Criterion 1 requires inspection and test programs to be defined and in place before reactor startup to ensure the integrity and operability of piping systems and equipment.

WSRC issued a report ("Restart Inspection and Testing Plan," Reference 8) to address this criterion. The Restart Inspection Plan provides information on the following:

- o process water piping
- o periodic equipment inspection-schedule (DPSOL-105-1851B, Reference 9)
- o ultrasonic testing of the reactor tank
- o Westinghouse Independent Safety Review (WISR) and Reactor Operations Management Plan (ROMP) items

The Restart Inspection and Testing Plan also outlines the post-restart ISI plan and describes (1) responsibilities and organization, (2) repair and replacement, and (3) program implementation.

The staff evaluated the Pump and Valve Testing Restart Plan and determined that WSRC has addressed those systems important to safety and identified pumps and valves that should be tested and the types of testing that must be performed.

14.3.2 Restart Issue Management

Use of Certified Inspectors (QA-2.1)

The WSRC Restart Quality Assurance group has performed a surveillance of the construction quality verification activities to ensure that independent inspections are performed by inspectors who are certified in their appropriate disciplines. In addition, WSRC's Environment Safety and Health Quality Assurance (ES&H QA) Division (QA Department) has verified that the independent inspections performed by Restart Quality Assurance have been performed by inspectors certified in appropriate disciplines.

WSRC Quality Assurance Manual 1Q (Reference 10) procedure QAP 2-5, Revision 0, provides the detailed requirements for compliance to DOE Order SR 5700.6C and NQA-1. This procedure specifies requirements and responsibilities for training, qualification, and certification of those WSRC independent inspection personnel who verify compliance of items, products, and activities to specified requirements. The procedure states that cognizant WSRC Quality Function (CQF) management is to perform an evaluation of personnel qualifications to determine the extent of formal and on-the-job training necessary to meet the independent inspector requirement. Formal core training programs for the qualification of independent inspectors are to be established and documented by the Site Quality.

The procedure establishes three certification levels for independent inspection personnel and the qualification requirements for each level. The limits for each level are related to functional activities rather than organizational position. Re-evaluations are conducted at periodic intervals not to exceed 3 years.

Use of Certified NDE Personnel (QA-2.2)

WSRC QA Plan (Reference 26) Section 9.0, and QA Manual, Procedure QAP 9-1, properly address the requirement of QA-2.2. The QA Plan also addresses conformance to the standards of American Society of Nondestructive testing SNT-TC-1A (Reference 11) for qualification and certification of NDE personnel.

Use of Approved Procedures for NDE Activities (QA-9.5)

NDE activities conducted on restart safety systems are required to be performed to approved procedures that clearly specify or reference the acceptance criteria for the activities to be performed. WSRC QA Manual 1Q, Procedure QAP 5-1, Revision 0, addresses this requirement.

WISR-6 requires a surveillance program to monitor potential flow degradation in the sump pumps and in the pumps in the ECS and CWS. This is addressed in Section 14.3.4 below.

WISR-8 requires a valve inspection program which includes a list of valves prioritized for inspection on the basis of the importance of their function to safe operation of reactor systems. This is addressed in Section 14.3.4 below.

(3) In-Service Inspections (ISI)

An In-Service Inspection (ISI) program shall be in place prior to restart, consisting of the following :

Periodic Equipment Inspections -- Implementation of this program shall include inspections as necessary to monitor component integrity and the functional capabilities of equipment.

Other Restart Issues -- Issue DB-6.4. (Charlotte Criteria, Reference 2) requires review of the adequacy of technical information on vessel integrity status, including results of previous inspections. Ultrasonic testing (UT) of the reactor tanks shall be performed before reactor restart. Inspection acceptance criteria shall be per WSRC-RP-89-208 (Reference 3), as approved by DOE.

The adequacy of the existing program shall be justified. One means of accomplishing this would be to compare the existing ISI program to the ISI requirements of the Section XI of the ASME Code (Reference 4).

(4) In-Service Testing (IST) of Pumps and Valves

In-service testing of pumps and valves shall be performed according to the following commitments prior to restart:

Periodic equipment tests and replacements shall be performed for the safety related pumps and valves. This shall be a prioritized program based on the importance of the equipment function to reactor operation.

Motor-Operated Valve Analysis Testing Systems (MOVATS) shall be established. WSRC shall develop a list of critical valves to be tested under the MOVATS program before restart. That list shall be submitted to DOE for review and approval.

The adequacy of the existing program shall be justified. One means of accomplishing this would be to compare the existing IST program to the IST requirements of the Section XI of the ASME Code.

The requirements of WISR-6 and WISR-8 shall be met.

(5) Qualification of Nondestructive Examination (NDE) and Test Personnel

Personnel conducting the ISI/IST and their acceptance shall be trained and qualified as specified in NQA-1, as required by DOE Orders SR 5700.6C (Reference 5). Qualification of NDE personnel shall also be in accordance with SNT-TC-1A and Section XI of the ASME Boiler and Pressure Vessel Code, as applicable.

(6) ISI and IST Post-Restart Programs

ISI and IST program shall be developed in accordance with the Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP), NUREG-0800,

14 INSPECTION AND TESTING

14.1 Background

The inspection and testing program for the Savannah River Site (SRS) reactors is essential to ensure that the operability of the systems, components, and structures will be demonstrated during normal and credible abnormal reactor conditions. The existing scheduled inspection and testing program for the SRS reactor systems and components include the performance of routine surveillance and maintenance work. Therefore, the current inspection and testing program must be revised and expanded to cover the whole scope of inspection and testing criteria. The Operating Contractor, Westinghouse Savannah River Corporation (WSRC), has committed to upgrade the SRS inspection and testing programs to acceptable standards that are comparable to those of commercial nuclear power plant.

14.2 Restart Criteria

The criteria for determining the adequacy of the SRS inspection and testing program for restart are described below.

(1) Scope and Issue Definition

Inspection and test programs shall be defined and in place before reactor start up to ensure integrity and operability of piping systems and equipment. The Inspection and Test Plan shall delineate the scope of each program including procedures, frequencies, quantities, extent, acceptance criteria, and applicable codes and standards. The Inspection and Test Plan shall be reviewed and approved by DOE.

(2) Restart Issue Management

Restart issues shall be resolved prior to reactor restart. The following items shall be among those Reactor Operations Management Plan (ROMP) issues (Reference 1) to be addressed by the Inspection and Testing Program.

QA-2.1 -- Use of certified inspectors

QA-2.2 -- Use of certified nondestructive examination (NDE) personnel

QA-9.5 -- Use of approved procedures for NDE activities

WISR-6 -- Pump test and monitoring for degradation (selected list)

WISR-8 -- Valve inspection and evaluation

WISR-38 -- Surveillance of Cooling Water System (CWS) piping

25. WSRC, R-3.2 (DPSOP 338), "Reactor Maintenance Administrative Procedures," Rev. 7, October 1990.
26. WSRC, DPSOL 105-1851E, "Test Following Non-Routine or Emergency Repairs," Rev. 1, September 13, 1989.
27. WSRC, ROSI II.12, "Test Control," Rev. 3, June 14, 1990.
28. WSRC, ROSI IV.11, "Control of Work," Rev. 3, June 25, 1990.
29. WSRC, ROSI IV.14, "Control of System Status," Rev. 0, December 1, 1989.
30. WSRC, "Conduct of Operations Manual," Rev. 0, May 11, 1990.
31. WSRC, DPSOL 338-211-1, "Test Control," Rev. 1, January 7, 1988.
32. WSRC, RP 3.103, "Post-Maintenance Testing Procedure," Rev. 0, April 8, 1991.
33. WSRC, DPSOL 338-204-1, "Procurement Document Control," Rev. 1, January 7, 1988.
34. WSRC, MRP 3.24, "Purchase Requisitioning," Rev. 0, February 16, 1990.
35. WSRC, DPSOL 338-208-1, "Identification and Control of Items," Rev. 2, April 1, 1990.
36. WSRC, DPSOL 338-207-3, "Supplier Evaluation Program," Rev. 1, January 7, 1988.
37. WSRC, RDP 8.01, "Control of Nonconforming Items," Rev. 1C, November 15, 1990.
38. WSRC, DPSOL 338-212-1, "Control of Measuring & Test Equipment," Rev. 2, September 28, 1990.
39. WSRC, DMP 3.05, "Control of Process Instrumentation," Rev. 0, September 1, 1990.
40. WSRC, RPSP 101, "Management Monitoring," Rev. 2, February 1, 1990.
41. WSRC, DPSOL 338-218-4, "Reactor Maintenance Management Inspections," Rev. 0, August 1, 1990.

5. _____, INPO 85-032, "Preventive Maintenance," Rev. 1, December, 1988.
6. WSRC, "Westinghouse Savannah River Company Quality Assurance Manual 1Q," Rev. 0, April 1, 1990. (Replaces DPW-83-111-3).
7. WSRC, RP-3.1002, "Conduct of Maintenance," Rev. 0, April 8, 1991.
8. WSRC, RD-3.1/DM-3.1, "RRD Work Manual," Rev. 0, April 1990.
9. WSRC, DPSOL 338-213-1, "Handling, Storage, and Shipping," Rev. 1, January 7, 1988.
10. WSRC, RTR-2634, "Review Plan for Analysis of Work Backlog," (Issue MS-15), Rev. 3, November 14, 1989.
11. WSRC, RTR-2667, "Operational Readiness Review Work Backlog, Corrective and Preventive Maintenance," Rev. 1, July 19, 1989.
12. WSRC-TR-90-42-133, "Uniform Listing of Reactor Systems, Acronyms, and Safety Systems", December 13, 1990.
13. WSRC, DPSOL-105-6101A, "Sample Analysis Schedule and Control Routine."
14. WSRC, DPSOL-105-6101B, "Sample Description and Location."
15. WSRC, DPSOL-105-6101C, "Sample Analysis Schedule and Control Data Sheet."
16. WSRC, DPSOL-105-1108, "-40 Level Data Sheets."
17. WSRC, DPSOL-1851A-PLK, "Periodic Equipment Test and Replacements-Schedule and Record," Rev. 19, January 2, 1990.
18. WSRC, DPSOL-1851B-PLK, "Periodic Equipment Inspections-Schedule," Rev. 14, January 10, 1990.
19. WSRC, DPSOL-1851C-PLK, "Schedule for Periodic Cable Inspections," Rev. 7, January 5, 1990.
20. WSRC, K-Area Standing Instruction D-70, DPSOL Schedule and Compliance, Rev. 0, February 13, 1991.
21. WSRC, OPS-ROTP-91167, "Safety-Related Procedures List," June 26, 1990.
22. WSRC, DPSP-85-1090, "Procedure Writer's Guide," Rev. 9, January 1, 1990.
23. WSRC, DPSOL-338-205-1, "Instructions, Procedures and Drawings," Rev. 2, March 21, 1990.
24. WSRC, K-Area Standing Instruction D-62, Restart Work Screening, Rev. 2, March 6, 1991.

of the system. These indicators provide input that can be used to identify problems directly affecting the reliability and operability of equipment. Pending the development of a near-term process to track and analyze recurring maintenance activities, this remains an open item.

13.4 Conclusion

The maintenance self-assessment was an excellent evaluation of the state of the maintenance in the reactor areas. The MIP established some ambitious corrective actions to upgrade and establish a good all around maintenance program. The MIP was based on available resources and budget as well as on the importance of the issue to restart requirements. Due to the importance of restart activities and the larger than expected drain on available resources to control and implement those activities, certain areas of the MIP have not proceeded as scheduled. Other maintenance areas addressed by the MIP have undergone temporary enhancements to correct recognized deficiencies that were not part of the original MIP but were needed in support of the restart effort. Some areas of the MIP have been harder to implement than originally planned and scheduled (i.e., post-maintenance testing and conduct of maintenance).

Overall the maintenance improvement effort is proceeding in a direction to support restart. This direction is not always in strict agreement with the MIP. The maintenance effort is being modified as restart nears and the actual needs and mechanics of the situations in the reactor areas are better understood. The structure of a good maintenance organization is developing. This is not to imply that the full development and implementation of the programs addressing the self-assessment corrective actions are not required or should not receive appropriate management attention. The identified actions of the MIP are required to bring the maintenance program to maturity and institutionalize the enhancements and initial changes being made for restart.

The staff has identified a number of open items in the maintenance program as discussed above. Upon satisfactory resolution of these open items, the staff will be able to conclude that the maintenance programs are acceptable for restart. The resolution of those open items will be addressed in a supplement to the SER.

13.5 References

1. Institute of Nuclear Power Operations, INPO 85-038, Guidelines for the Conduct of Maintenance at Nuclear Power Stations, Rev.0, October 1985.
2. WSRC, OPS-RMK #900334, "Reactor Maintenance Self-Assessment Report," Rev. 0, January 29, 1990.
3. WSRC, OPS-RMK #900580, "Reactor Maintenance Improvement Plan Report," Rev. 0, April 27, 1990.
4. Institute of Nuclear Power Operations, INPO 85-001, "Performance Objectives and Criteria for Operating and Near-Term Operating License Plants," Rev. 1, April 1990.

the program is delayed until implementation of the problem analysis program in April 1992. MIP Section VII implements the work control program, which has provided a method for maintenance history documentation. This section of the MIP was implemented April 2, 1990, and will provide the data that will be used to fill the maintenance history data base. MIP Section XVIII addresses the need to upgrade the Work Management System (WMS) to allow input of maintenance history data. This section of the MIP is scheduled for implementation in October 1991 and supports the scheduled development of the problem analysis program.

The Work Control Manual (RD-3.1) provides the directions and the mechanism for the recording and retention of maintenance activity documentation. Continued effort is required to ensure compliance with the intent of RD-3.1. On the basis of its review, the staff finds that WSRC has met the requirements of this section for development of a plan and schedule for the reactor maintenance history program. The implementation of this program will be reviewed and discussed in a monthly operations report.

Analysis of Maintenance Problems

For those systems or equipment that are important to safety or safety related, a mechanism to identify and track reoccurring maintenance problems should exist. This program should provide guidance and evaluation of these problems to ensure timely action to correct reoccurring maintenance problems and prevent rework.

WSRC evaluated the programs for performing the systematic analysis of root causes behind unplanned occurrences against the INPO 85-038 criteria. The results of the evaluation are included in the Reactor Maintenance Self-Assessment Report. A plan and schedule to address self-assessment corrective actions, including training, are included in MIP Section II, "Maintenance Problem Analysis Program." The MIP includes a provision for developing a program that provides guidance and training to ensure that unidentified causes of failures are evaluated and understood. The program shall include provisions for information collection, analysis, cause determination, corrective action, and follow-up. The MIP schedules the implementation of this program for April 1992. Existing maintenance procedures and instructions include the need to record information that might help in root cause analysis as well as direct that an attempt be made to determine the root cause of a malfunction if possible. RD-3.1 directs the implementing work group supervisor to attempt to determine the root cause of the failure during repair.

The Reactor QA group monitors NCRs for generic and repeat problems. The reactor quality assurance (RQA) process will reflect reoccurring programmatic type issues but has not been set up to monitor and assess repeat maintenance activity.

The staff was unable to identify a near-term process or mechanism for the identification and analysis of recurring maintenance items with respect to important-to-safety equipment. These issues may be indicators of problems inherent in the design of the equipment, operation, training or in the design

Management Involvement

Corporate and facility managers should be technically informed and personally familiar with the conditions at the operating station. The program needs to effectively monitor the day-to-day performance of maintenance activities to identify the areas where assets and resources can be best utilized. They should review and be familiar with the performance of maintenance activities to verify that they are being effectively accomplished within established guidelines. They should use their information to effect changes and upgrade the programs and training to support the effective and efficient performance of maintenance activities in a timely manner. Mechanisms for the correction and tracking of program and material deficiencies need to show proper evaluation, prioritization, documentation, and tracking.

Several areas of the MIP address the self-assessment corrective actions. Two WSRC management inspection programs have been implemented to provide management involvement in the maintenance process. These programs, in conjunction with the implementation of RD.3.1 (Work Control Manual) and RP 3.1002, "Conduct of Maintenance," address most of the self-assessment identified corrective actions. Management monitoring at the senior management level is performed per the directions of RPSP 101, "Management Monitoring" (Reference 40). At a lower level, Reactor Maintenance Department management inspection requirements are addressed by DPSOL 338-218-4, "Reactor Maintenance Management Inspections" (Reference 41). These procedures provide instructions that define the requirements for management inspections including (1) who is to perform the inspections, (2) inspection frequency, (3) inspection duration, (4) methods of reporting inspection findings, and (5) a tracking, resolution, and follow-up process for the findings. Prior to the implementation of the management inspection program, training based on INPO's observation training program was provided to all levels of Reactor Maintenance supervision. Long-term corrective actions of the self-assessment are addressed by the MIP. On the basis of its review, the staff finds that the requirements of this section of the restart criteria are met.

Maintenance History

The maintenance history program should clearly define the systems and equipment that require documentation and retention of historical data. There is a need to control and maintain maintenance records to provide a system to allow retrieval of maintenance records and equipment information that results from maintenance activities. These records will include all the information required to support and document the required condition of systems and equipment through restart. Management should ensure that maintenance history is recorded on all completed work documents.

WSRC evaluated the reactor maintenance history program against the criteria of INPO 85-038. The results of the evaluation are included in the Reactor Maintenance Self-Assessment Report. The MIP provides a schedule and plan to implement the corrective actions of the self-assessment. The MIP divides the program for maintenance history into three areas based on the required corrective action. MIP Section II establishes an equipment tracking and trending program. As trending is part of problem analysis, implementation of

Maintenance Tools and Equipment

The availability of the appropriate tools and equipment directly affects the ability of the maintenance department to maintain the plant in a safe and reliable condition. WSRC evaluated the policies and procedures that control storing, using, and maintaining maintenance tools and equipment against the criteria of INPO 85-038. The results of the evaluation are included in the Reactor Maintenance Self-Assessment Report. The restart criteria allow this requirement to be addressed by the Maintenance Improvement Program. The safety-related aspects of tool and equipment control are addressed by the M&TE programs in DPSOL 338-212-1 and RD-3.1. MIP Sections VIII, X, XIII, and XX include a schedule for implementation of the self-assessment corrective actions related to maintenance tool and equipment control throughout 1991. DOE has reviewed the proposed corrective actions and their implementation schedule and determined them to be acceptable.

Station Material Condition Inspection

The involvement of facility management in periodic station walkdowns and inspections improves and maintains the material condition of the facility and clearly displays management standards to all personnel. The ability to identify, track and correct facility material deficiencies is an important step in maintaining facility equipment in a condition of maximum safety, reliability, and availability. There is need to provide training, instruction, and example to plant personnel to emphasize the importance of plant material condition. Personnel need to be trained to recognize material condition deficiencies and the expected actions to be taken when material condition deficiencies are identified. Management should reinforce these expected actions via management involvement or inspection activities.

WSRC evaluated the station material condition policies and procedures against the criteria of INPO 85-038. The results of the evaluation are included in the Reactor Maintenance Self-Assessment Report. MIP Sections III and XII includes a plan and schedule to implement a maintenance inspection program and a conduct of maintenance program prior to restart. These two programs address 14 of the 15 self-assessment corrective actions that are related to station material condition. The one remaining corrective action is addressed by MIP Section XIV, "Maintenance Interface and Support Activities." This MIP item addresses the need to monitor, control, and document plant modifications and system changes and is scheduled for implementation in 1991.

Training of facility maintenance personnel to emphasize the importance of plant material condition has not been fully completed. RD 3.1 provides instructions directing the initiation of a work request when deficiencies are observed. Training to implement RD 3.1 was completed in April 1990. The conduct of maintenance procedure reinforces the required attitude and expected performance standards of maintenance personnel as related to the identification of deficiencies. Training to implement the conduct of maintenance procedure has not been completed. This is an open item which is tracked under maintenance "Training," as previously discussed.

DPSOL 338-212-1 (Reference 38) provides controls and the general calibration requirements for M&TE. RD-3.1 requires maintenance planning to include within the work package the specific tools necessary to support the implementation of the proposed maintenance activity. DPSOL 338-212-1 provides the administrative controls governing the use of M&TE, storage location, and recalibration frequency and the process to be implemented for M&TE that is identified to be out of calibration. This process includes an NCR and followup disposition for each item on the appropriate Instrument/Tool Report (ITR). The ITR and the M&TE Sign-Out Log document each use of a particular piece of M&TE. Adequate control of M&TE should include requirements for selection of M&TE to ensure that the proper piece of M&TE is used in any specific maintenance activity. DPSOL 338-212-1 requires the use of M&TE for the collection of quantitative data to have an accuracy and range equal to or greater than the required tolerances of the equipment being measured. All revised maintenance procedures include specific M&TE requirements. Work Instructions, including the required values and tolerances, are supplied by the Reactor Engineering during the work package approval cycle per the requirements of RD-3.1. The work planning group then selects the M&TE required for the activities being performed. The selection of M&TE is based on good engineering judgement and on what the Reactor Quality Verification group will accept. DMP 3.05, "Control of Process Instrumentation" (Reference 39), specifies the accuracy requirements for M&TE used to calibrate process instruments. If required to ensure each piece of M&TE is operating properly, M&TE can be functionally checked prior to use. DPSOL 338-212-1 specifies that, if called out by the procedure or specific specification in use, a calibration verification is to be performed. The type of calibration or check is to be specified by the procedure or specification calling for the calibration verification. The functional check prevents the use of M&TE that needs recalibration or would otherwise provide false indication or results. Special M&TE may be provided by vendors for use in maintenance activities.

The requirement for the procurement of vendor calibration services when used to calibrate Process Instrumentation has been identified and addressed in DMP 3.05, MRP 3.24, and DPSOL 338-204-1. Through the procurement process, the requirements for the control of this equipment can be made equivalent to Reactor Maintenance Department supplied M&TE. Segregating M&TE that has been identified as defective or is out of calibration is done through the use of RDP 8.01, "Control of Nonconforming Items." The NCR procedure requires that those conditions that are adverse to quality be indicated with an NCR tag. DPSOL 338-212-1 requires that deficient or uncalibrated M&TE be evaluated under the NCR process and identified with the appropriate QA hold tag.

The DOE staff has determined that the M&TE program does not fully meet the restart criteria in that the following aspects of M&TE use and control have not been addressed. The DPSOLs and RD-3.1 have not provided guidance or instructions for the controls needed to prevent contamination of M&TE or identified facilities for the calibration of contaminated M&TE. DPSOL 338-212-1 does not provide for a master list of M&TE or provisions to recall M&TE prior to the expiration of calibration. No requirements were identified that addressed the need to calibrate M&TE prior to being discarded. Pending a revision of DPSOL 338-212-1 or other procedures to include the appropriate requirements, this issue remains an open item.

Reactor Maintenance in an important-to-safety system must be evaluated prior to installation. The Responsibilities Section of DPSOL 338-213-1 requires that the Reactor Maintenance Department mechanics maintain appropriate storage requirements and comply with any specific handling requirements for maintenance-related spare parts and materials. The above DPSOLs, as implemented by RD-3.1, provide a process to ensure that the parts and materials used during safety-related maintenance activities meet and are handled in accordance with any known design and quality requirements. RDP 8.01, "Control of Nonconforming Items," provides the mechanism to address the loss of traceability, the dedication, or other indeterminate condition of a part or material that Reactor Maintenance receives from stores. MIP Section XV addresses long-term corrective actions related to materials management issues as identified by the maintenance self-assessment.

MRP 3.24 addresses the need and provides a mechanism to define the requirements associated with the quality control requirements of parts, materials, and services. Reactor Maintenance Administrative Procedures (DPSOL 338 series) implement the basic requirements of QA Manual IQ relating to the handling of spare parts and materials. On the basis of the procedures reviewed as discussed above the staff finds that WSRC has established adequate procedural guidelines for proper control of materials which are to be used in reactor areas.

Control and Calibration of Measuring and Test Equipment

For M&TE used to support maintenance and test activities on important-to-safety and safety-related systems and equipment, the M&TE program should ensure the accurate performance of station instrumentation and equipment for testing, calibration, and repairs. M&TE includes all devices or systems used to inspect, test, calibrate, measure, or troubleshoot in order to control or acquire data verifying the conformance of an instrument or piece of equipment to specified requirements. M&TE selected for use should have the precision necessary to ensure that station instrumentation and equipment will operate within the design accuracy requirements. All M&TE should have a unique identification number that accurately identifies the specific device and provides positive traceability. The M&TE calibration program must be based on standards that are traceable to a national standard or are recognized standards themselves. Procedures should be used to calibrate M&TE to help control the performance of the calibrations, provide repeatable calibrations, and provide approved acceptance criteria. M&TE should be functionally checked when applicable. Facilities should be provided to control storage, issue, and calibration of M&TE. M&TE with suspected or actual deficiencies should be segregated and marked to prohibit its use. M&TE that is not fully calibrated or useable should be clearly marked to indicate its limitations. M&TE issue should include records for accountability and traceability of use. M&TE found out of calibration or defective should receive a timely evaluation to determine the validity of all calibrations for which it was used. Reviews to determine that the control of M&TE is effective should be periodically conducted.

DPSOP-338-208-1 (Reference 35) provides requirements for the identification and control of procured items. The process for the procurement and validation of vendor and contractor services is provided by the procedures that outline the procurement process and is verified by a process provided in DPSOL 338-207-3, "Supplier Evaluation Program" (Reference 36). The requirements for the proper storage of materials is initiated by DPSOL 338-204-1, which specifies storage requirements for procurement purposes. DPSOL 338-208-1 addresses the applicable identification and class storage requirements to implement the procurement specifications. The DPSOLs and RD 3.1 implement the basic requirements of QA Manual, 1Q, related to procurement for safety-related applications.

The process for the qualifying of deficient and nonconforming parts and the substitution of parts is not directly addressed by the procurement process. If a part that does not meet the requirements of original design is to be used, it must be addressed by a nonconformance report. The NCR process, RDP 8.01, "Control of Nonconforming Items" (Reference 37), provides the mechanism through the use of repair, rework, use-as-is, and reject dispositions to dedicate the part or reject the part as necessary. This issue remains an open item pending the receipt and review of the plan that will address the remaining procurement related corrective actions.

Material Receipt, Inspection, Handling, Storage, Retrieval, and Issuance

As related to the parts that support safety-related or important-to-safety equipment, maintenance and stores personnel must be aware of the correct process to receive, inspect, handle, and store station material and equipment so that it is easily retrievable and usable when it is issued. These policies must be understood and implemented by maintenance personnel and other groups that interface with the stores operations. Procedures must specifically describe the responsibilities and techniques for receiving, inspecting, handling, storing, retrieving, and issuing materials. For parts and materials designated for safety-related or important-to-safety systems, the quality of the items received must be in accordance with the applicable specifications and maintained, issued and installed consistent with those specifications.

DPSOLs 338-204-1, 338-208-1, and 338-213-1 are used to meet the basic requirements for the receipt, storage, inspection, and handling of material as defined by the QA Manual. These documents define Reactor Maintenance Department responsibilities relating to the receipt, storage, inspection, and handling of maintenance materials. To ensure adequate control of maintenance parts and materials, Reactor Maintenance has established specific end user controls. DPSOL 338-208-1 provides the requirements for the identification and control of maintenance materials and parts. DPSOL 338-213-1 provides requirements and directions for the packaging and storage of maintenance materials and parts while they are in the custody of the Reactor Maintenance Department. Section 4.0 of DPSOL 338-213-1 provides control and directions for in-storage preventive maintenance. MRP 3.24 requires that Controlled Items (Level 1) be receipt-inspected by QC. DPSOL 338-204-1 requires the Reactor Maintenance Department to inspect all Level 2 and 3 procured items to ensure they are as specified by the procurement order and are not damaged. DPSOL 338-208-1 requires that all spare parts or machinery installed by

extent of testing required. Specific guidance governing the criteria and methods to be used to ensure that the maintenance activity is adequate and the equipment is functioning properly has not been provided. The Work Control Manual has identified the need for acceptance criteria, but RD-3.1 has not included any specific guidance that can be used during the development of work instructions if a DPSOL or routine test is not used. A Post-Maintenance Testing Procedure, RP 3.103 (Reference 32), has been approved but has not been fully implemented. This procedure provides the structure, format, and approval process for the creation of PMT matrices, but none of the matrices have been created. These matrices should provide the guidance for actions to be completed from the maintenance perspective to verify that equipment will perform its intended function after maintenance. Pending creation of the appropriate PMT matrices and the implementation of the PMT procedure, this issue remains an open item.

Procurement of Parts, Materials, and Services

An effective procurement process to ensure that the proper parts, materials, and services are available when needed must be in place for restart. Policies should be established for the procurement of parts, materials, and services that are used in maintenance activities on important-to-safety and safety-related systems. Quality assurance procurement records need to be controlled and maintained to provide documentation for qualified parts and materials and ensure traceability of parts and materials. Deficient or nonconforming items should be resolved in an effective and timely manner. A process should be provided to qualify non-qualified material. A process of acceptable substitution should be provided to obtain parts that are no longer available. Design requirements should be reviewed by appropriate personnel to ensure the upgraded or substitute parts are consistent with the application of the part and component. Retrievable documentation should exist to support the identification of inspection and testing requirements necessary to ensure the qualification and acceptability of the part. Personnel must understand their responsibilities in regard to the implementation of the requirements of the procurement program and policies.

WSRC evaluated the procurement of parts, materials, and services against the criteria of INPO 85-038. The results of the evaluation are included in the Reactor Maintenance Self-Assessment Report. A plan and schedule to address self-assessment corrective actions is included in the MIP. Section VII of the MIP implemented the Work Control Manual (RD-3.1). RD-3.1 implements the self-assessment corrective actions for the identification of long lead time procurement items as well as the identification, procurement, control, and storing of parts and materials for maintenance activities. Section XV of the MIP provides a schedule for the development of a program plan that will address the remaining procurement-related corrective actions. The program plan and its associated implementation schedule are due prior to restart. There are existing programs that have criteria for the procurement of parts, materials, and services and address safety-related requirements. DPSOL 338-204-1 (Reference 33) establishes responsibilities and prescribes the procedural and administrative requirements for creating a stores requirements document. MRP 3.24 (Reference 34) defines the requirements for the preparation, review, approval, and control of procurement requisitions.

each department's responsibilities, equipment to be included, degree and type of testing, procedure needs, acceptance requirements, testing control, and results documentation.

The need to implement post-maintenance test (PMT) activities is included in the instructions of the Work Control Manual. Section 4.23, "Performance Testing of the Work Control Manual," dictates the inclusion of steps or procedures in the work package for the performance of test steps. The format could take many forms, from instructions within the work document to Special Test and one-time only DPSOLs. The format depends on the maintenance activity and the type of testing required. Reactor Operations Standing Instructions provide a mechanism that can be used to conduct test activities. DPSOL 105-1851E "Test Following Non-routine or Emergency Repairs" (Reference 26), is used with a work package, as directed by ROSI II.12 "Test Control" (Reference 27), to implement testing of equipment, when necessary, to perform an operability test following non-routine or emergency repairs. For integrated functional or operability testing, ROSI II.12 provides direction for including acceptance criteria in the test document generated per the format of DPSOL 105-1851E. DPSOL 105-1851E is included within the RD-3.1 work instructions and is used when performing the necessary testing. RD-3.1 Sections 3.0, "Responsibilities," and Section 4.12, "Work Package Preparation," identify the requirement to include acceptance requirements in the work package. Test control, from the Operations perspective, includes controls for equipment and testing as established by ROSI IV.11, "Control of Work" (Reference 28), ROSI IV.14, "Control of System Status," (Reference 29) and "K-Area Standing Instruction D-56." These instructions govern the operational aspect of the control of work and the control of equipment status.

Additional guidance on the operation and precautions to be exercised by the operators during testing and equipment startup activities is included in the Conduct of Operations Manual (Reference 30). Section 7.D of the Conduct of Operations Manual elaborates on the requirements of the ROSI II.12 as they specifically relate to PMT. Test controls, from the Reactor Maintenance perspective, are included in DPSOL 338-211-1 (Reference 31), which provides general instructions and control for the Reactor Maintenance Department. The DPSOL includes responsibilities concerning test control and the qualification of personnel. This instruction provides test controls for SPs, RSP, Special Test, "Test" DPSOLs, and non-routine Test DPSOLs. WSRC evaluated the PMT program the criteria of INPO 85-038 Guidelines for the Conduct of Maintenance at Nuclear Power Stations. The results of the evaluation are included in the Reactor Maintenance Self-Assessment Report. The Maintenance Improvement Program (MIP), Section IX, establishes a plan and schedule to implement the corrective actions identified in the self-assessment report.

The DOE staff determined that the post-maintenance testing (PMT) program does not fully meet the restart criteria in that certain aspects of the PMT program required by the restart criteria have not yet been implemented. No specific guidance has been provided that relates the maintenance activity to the importance of the equipment and to the type and amount of testing required. Both Reactor Maintenance and Reactor Operations procedures recognize the need to perform testing after the completion of maintenance activities. However, these procedures have not provided the necessary guidance to determine the

K-Area Standing Instruction D62, Restart Work Screening (Reference 24), has recently been implemented to define and provide directions on processing work requests. The information provided in the instruction supplements the requirements of RD 3.1, Work Control Manual. The instruction provides the routing and review requirements for each work request by priority. This instruction provides the structure and controls which ensure that preplanning and scheduling can take place.

Control of Maintenance Activities

Certain functions will be required to effectively implement the work control process. There is a need to establish a routine maintenance practice that requires maintenance personnel to inform reactor operations of any abnormality associated with the implementation of a maintenance procedure. The control of prints, manuals, and references is needed to allow for more effective and timely planning of corrective maintenance activities, as well as more effective, timely, accurate, and confident planning and execution of troubleshooting activities. Access to the most technically correct and controlled documents is needed on a routine and repeat basis for the performance of all maintenance activities. Implementation and revision (as necessary) of the new work control system is required to develop the required working relationship among all departments that support the maintenance function and ensure the integration of the maintenance functions with each department's programs and procedures. WSRC will be required to demonstrate that the maintenance engineering staff is effectively supporting day-to-day maintenance activities.

The Work Control Manual, RD-3.1, and R-3.2, "Reactor Maintenance Administrative Procedures" (Reference 25), implements a work control program based on the requirements identified in the Reactor Maintenance Self-Assessment. These manuals provide the necessary requirements and organizational responsibilities for the development, approval, and performance of maintenance activities.

No integrated program exists to ensure the availability and update of all the needed reference material, drawings, and manuals for planning and maintenance personnel. Pending the implementation of a process to provide the needed documents for routine maintenance activities, this remains an open item.

Post-Maintenance Testing

Post-maintenance testing includes all testing performed following maintenance activities. In the case of corrective maintenance to correct equipment problems, post maintenance testing determines if additional maintenance work or diagnostic fact finding should be performed. Equipment monitoring after repair during extended operation provides assurance that the cause has been properly corrected. Post-maintenance testing is performed in adequate detail to verify that the cause of a malfunction has been corrected and associated equipment has not been affected by the performance of maintenance. Post-maintenance testing applies to all maintenance activities and addresses

not been addressed by the PWG, but RD-3.1 identifies requirements for including testing requirements in work packages created under the work control program. Although not specifically included in the applicable maintenance procedure, the requirement to include the retest requirements in the work package addresses this issue. The procedure upgrade effort for those procedures that have been identified as required for restart have been scheduled with the appropriate reports on progress being performed weekly.

The procurement process includes the controls needed to evaluate and ensure that vendor procedures meet the requirements for use within the facility. The content, format, structure, review and approval of maintenance procedures, as outlined by the PWG, address all of the issues related to maintenance procedure content that were evaluated above. RD 3.1, DPSOL 338-205-1, and document control procedures provide a mechanism to obtain and control drawings and procedures required in the performance of maintenance activities. Included in these procedures are requirements for the maintenance personnel to work only from the most recently approved and issued documents including any applicable approved changes. DPSOL 338-205-1 (Reference 23) provides a requirement for the biannual review of maintenance procedures. An engineering and QA review process has been provided to use procedures that have not been through a biannual review on a one-time-only basis. Procedure writers have been trained to the requirements of the PWG, but a certification program has not been completed for those personnel who are performing the walkdowns and validation. An open item identified in SER Section 10.3 documents the lack of a validation certifications program. The process for the review and approval of maintenance procedures ensures that design requirements have been included. If information is required during the writing of a procedure, a Request for Information (RFI) is submitted to engineering requesting the needed information.

A formal means to track and identify commitments in the procedures is required to ensure that adequate procedures are available and used. Procedural requirements related to Technical Specifications are tracked by parenthetically noting the Technical Specification section that is applicable. This is a requirement of the PWG. No clear or consistent method of tracking commitments in maintenance procedures was identified. Procedure commitment tracking system is an open item in SER Section 10.3.

Planning, Scheduling, and Coordinating Maintenance

Several planning and scheduling groups have been added to the facility personnel to support the restart effort. These groups will be evaluated to ensure that they can effectively plan, schedule, and coordinate maintenance activities in support of equipment availability and operability requirements.

Through the use of WMS and planning program P-2, the maintenance and planning organization has a mechanism for tracking the status of maintenance activities. Additionally, WMS provides a tracking mechanism for preventive maintenance due dates. Work activities identified as necessary to achieve restart are tracked against a restart schedule.

Pending the review, identification, documentation, and means to control and implement a PM and surveillance tracking program for safety-related and important-to-safety equipment, this issue remains an open item.

No process has been identified for analyzing and trending data for all important-to-safety lube oil systems to detect equipment degradation. Pending a review and application, as required, of the lube-oil sampling program to all appropriate important-to-safety systems, this remains an open item.

Maintenance Procedures

Useable maintenance procedures integrated with the knowledge, ability, training, and attitude of the maintenance personnel must be available to achieve restart and ensure safe and reliable facility operation. There are aspects of maintenance procedures that directly relate to satisfactory performance of maintenance activities on important-to-safety and safety-related equipment. These procedures may also directly affect the control and operation of facility equipment and, therefore, the continued safe operation of the facility. Certain aspects of maintenance procedures as related to activity control and implementation will be evaluated to ensure that maintenance procedure steps that could cause a process transient are identified. Effective implementation of a validation program that consistently and adequately accounts for the integration of the maintenance activity with the control and operation of the facility is also required. The validation process must identify existing technical inaccuracies and ensure that the existing steps are clear enough to eliminate human errors. Administrative controls must include provisions to ensure that all procedures and references used are the latest revision and must establish requirements for the notification of operations personnel in a timely manner when procedural problems or other unusual events occur.

WSRC evaluated the maintenance procedures program against the criteria of INPO's Guidelines for the Conduct of Maintenance at Nuclear Power Stations. The results of the evaluation are included in the Reactor Maintenance Self-Assessment Report. The Maintenance Improvement Program (MIP), Section XVI establishes a plan and schedule to fully implement, after restart, all the corrective actions identified in the self-assessment report.

MIP Section XVI establishes a schedule for the review and revision of the maintenance program and procedures. The program is scheduled for implementation prior to restart and will continue through restart. The Reactor Restart Division has issued OPS-ROTP-91167, "Safety-Related Procedures List" (Reference 21), which contains a listing of all procedures required for review and revision prior to restart. This document provides the criteria used by Reactor Training and Procedures (RTAP) to evaluate the procedures. Tables 11 and 12 of this document provide the list of safety-related maintenance procedures that are a result of this evaluation. The RTAP review scope included an evaluation of those systems identified in SER section 5.6, Table 1. The procedure revisions are being conducted per the requirements of DPSP-85-1090, the "Procedure Writer's Guide (PWG)" (Reference 22). Section 3.1 of the PWG provides directions for the inclusion of return-to-service requirements, M&TE requirements, and QA hold points. Retest requirements have

Programs and procedures exist that identify surveillance and preventive maintenance requirements (i.e., WMS, DPSOL 1851A, B, and C [References 17, 18 and 19]). Due to the lack of integration and a documented basis for the preventive maintenance activities (PMs) and the lack of consistently identified requirements for surveillances, these programs have not been effectively implemented. Coordinators have been established for the tracking and initiation of DPSOL 1925, 1926, and 1851 series requirements. The requirements of these DPSOLs have been evaluated by engineering for safety considerations. The results of the evaluation are reflected in the tracking database for 1851 series activities. Technical Specification surveillance requirements are tracked by an Operations group. In accordance with the requirements of K-Area Standing Instruction D-70, DPSOL Schedule and Compliance (Reference 20), this group identifies on a daily basis all surveillance requirements to shift supervision. They also verify on a daily basis the completion or LCO tracking of due, complete, incomplete, and unsatisfactory surveillance results.

Separate coordinators have been established for the tracking and completion of 1851A, B, and C activities. The 1851 series of activities has been reviewed by Reactor Engineering for restart applicability. The results of the review are reflected in the tracking database. The importance of each activity is reflected in the database in terms of the need to repeat the activity at the procedural frequency, that is, required as an RTIR requirement, due again at the next outage, or due again at the indicated periodicity. The initiation of 1851A activities is done by the coordinator who identifies the requirement for the activity to the system planners. The planners then add the activity to the schedule for completion in a given time period. 1851B and C activities are initiated by the coordinators through work requests. These activities are then scheduled through the work control process. The 1851 coordinators must track and access the completion of the activity by monitoring the completion of the scheduled activities. Once the activity is determined to be completed, the coordinator can then access the document control files or the work history files to verify completion. This process is being effectively implemented to ensure the surveillances and other identified periodic activities are completed. This process is heavily loaded with scheduling and tracking personnel to achieve restart.

The preventive maintenance program must provide assurance that equipment and system functions are maintained for all operational and accident scenarios. Reactor Maintenance, in conjunction with Engineering, will evaluate, establish, and implement critical PMs for important-to-safety equipment. A process should be in place that reflects the results and basis of the PM review, as well as accurately reflecting the mechanism that is used to complete the PM and the means to call out and implement that mechanism at the appropriate frequencies.

There is a need to integrate the required maintenance and operational PMs and surveillances for tracking, consistency, and elimination of redundancy. The process to ensure that the Technical Specification surveillance requirements are properly identified and completed is under development. Performance of many periodic activities to meet these requirements will be the responsibility of Reactor Maintenance Department.

evaluate any corrective or preventive maintenance issues that have been identified since the cutoff date in RTR-2634. For major issues, the Restart Issues Management Program provides a mechanism and criteria to evaluate significant issues to ensure that restart and safety issues do not get overlooked. On May 10, 1990, the K-Area Reactor Plant Manager issued an unserialized memorandum implementing a preventive and corrective maintenance review process. The memo provides the process and the criteria for evaluation of emerging maintenance activities by K-Area Reactor Operations Department and support staff. The criteria are used in the evaluation process to categorize the maintenance activities that are required for restart and those not required for restart.

The WSRC Reactor Technology Department has issued a list of systems defined as "important to safety." WSRC-TR-90-42-133, "Uniform Listing of Reactor Systems, Acronyms, and Safety Systems" (Reference 12) defines the criteria for creating this list and includes the list of systems and equipment evaluated as important to safety. This list contains all the systems identified as safety-related in Section 5.6 of the SER. "Work Control Manual," RD-3.1 has established criteria and specific minimum requirements for maintenance activities on equipment or systems that are considered important to safety. These requirements include the use of Category 1 or 2 procedures or work instructions for the performance of the maintenance activities and the approval of the work package by Reactor Engineering and QA.

Reactor Operations Department routine rounds procedures are used to collect predictive maintenance information. Motor stator temperatures and bearing temperatures are recorded periodically on the operator's round sheets. Lube oil sampling for the Bingham pumps is performed using DPSOL 105-6101A, "Sample-Analysis Schedule and Control-Routine" (Reference 13); DPSOL 6101B, "Sample Description And Location" (Reference 14); and DPSOL 6101C, "Sample Analysis Schedule and Control-Data Sheet and Schedule" (Reference 15). Sampling is also called up periodically by DPSOL 105-1108, "-40 Level Data Sheets" (Reference 16).

A program has been developed to perform vibration analysis (i.e., Reactor Maintenance Memorandum #1011, Rev. 0, "Reactor Maintenance Machinery Vibration Monitoring Program, MVMP"). This document has been evaluated by the staff and is considered to be a valuable tool for both predictive and corrective maintenance. Motor inspection and testing is required by the WMS to be performed periodically in accordance with WI-E-MTR-02, the work instruction provided for normal maintenance and testing of motors.

WSRC evaluated the preventive maintenance program against the criteria of INPO's Guidelines for the Conduct of Maintenance at Nuclear Power Stations (INPO 85-038) and INPO 85-032, "Preventative Maintenance." The results of the evaluation are included in the Reactor Maintenance Self-Assessment Report. The Maintenance Improvement Plan (MIP), Section X establishes a plan and schedule to implement the corrective actions identified in the self-assessment report.

(E&I) mechanics and one for the Maintenance mechanics. The curriculum models reflect existing, as well as proposed, training modules. Priority for the training has been established by restart requirements and the specific maintenance training needs associated with scheduled restart maintenance activities and tasks.

Several new and revised programs governing and improving maintenance activities, as identified by the Maintenance Improvement Plan (MIP), are scheduled for implementation prior to restart. These programs are the result of corrective actions identified by the Reactor Maintenance Self-Assessment Report. The need for training is recognized on the MIP schedule prior to the implementation of the new program. The program areas scheduled by the MIP for implementation prior to restart are post-maintenance testing, conduct of maintenance, and the work control program. Training required to implement the new work control program has been completed as a restart issue.

The DOE staff has determined that maintenance training does not fully meet the restart criteria. The training programs for post-maintenance testing and conduct of maintenance have not been completed. Training is scheduled by the MIP for development and completion prior to the program implementation. Pending the completion of training activities for these new programs, this issue remains an open item.

Maintenance Facilities

"Work Control Manual," RD-3.1 (Reference 8), establishes the duties and responsibilities of a maintenance planning group, a K-Area scheduling group, and an outage control group. These groups are located in the K-Reactor area and coordinate the performance of maintenance activities within the K-Area. To support the maintenance effort required for restart, facilities to house and allow effective functioning of these groups has been recognized as a requirement, and additional facilities have been made available. Work and Shop facilities are modified as required to support the changing maintenance requirements and activities. An evaluation of maintenance facilities are included in the management inspection program, and a provision has been made to document and track the results of these evaluations and take corrective actions. RD-3.1, "Work Control Manual" and DPSOL 338-213-1, "Handling, Storage, and Shipping" (Reference 9), ensure tools, spare parts, and miscellaneous supplies located in the maintenance shops and the warehouse are stored correctly to maintain usability. The above adequately meets the restart criteria for facility maintenance.

Corrective, Predictive, and Preventive Maintenance

RTR-2634, "Review Plan for Analysis of Work Backlog, Corrective, and Preventive Maintenance" (Reference 10), describes and implements a program for the review of the backlog of preventive and corrective maintenance items through February 3, 1989. RTR-2667 "Operational Readiness Review Work Backlog, Corrective, and Preventive Maintenance" (Reference 11), which was issued as a result of the RTR-2634 review, identified the backlog of work items having safety impact that needed to be resolved prior to the restart of K-Reactor. The Restart Issues Management Program (RIMP) provides a means to

Without the implementation of these items, the maintenance program would be unable to implement safety-related aspects of the restart criteria. INPO 85-001 requires the development of maintenance standards that define the maintenance department objectives, expected performance levels, responsibilities, and accountabilities. These standards, as well as department goals, need to be communicated to the maintenance personnel through training, direct observation, and guidance.

Many maintenance performance objectives and criteria of INPO 85-001, Sections MA.1 through MA.10, are being implemented through existing procedures and instructions. Many more of the INPO 85-001 criteria are included in the corrective actions identified by the Maintenance Improvement Program (MIP). Implementation of the MIP-identified improvements are occurring or are scheduled to occur before, during, and after restart. The MIP has scheduled the implementation of a conduct of maintenance procedure prior to restart. The procedure, "Conduct of Maintenance Procedure," RP-3.1002 (Reference 7), has been issued, but requires that training be completed to be fully implemented. The procedure sets goals and establishes standards, objectives, expected performance levels, and responsibilities for maintenance activities and personnel. The approval of RP-3.1002 adequately addresses the criteria for restart. The MIP, Section VI, scheduled the development of a formal maintenance department goals and objectives program based on INPO 85-001 after restart. This program will address all the self-assessment-identified corrective actions. The staff considers this item to be closed.

Training

One-time only training was completed per the criteria of ROMP Section HP-8.1c to address many restart criteria areas. The lesson plans included areas relating to the control of combustibles, housekeeping, maintaining seismic integrity of the plant, professionalism, formality of maintenance operations, health protection, work control, and the tagout/lockout policy. Nonconformance report training is under development by the WSRC Site Quality Department. This area was not included in the HP-8.1c training.

The DOE staff has determined that restart training does not fully meet the restart criteria in that Reactor Restart Training did not address nonconformance reporting as delineated by ROMP HP-8.1c. Pending development and approval of the nonconformance report training lesson plan and completion of training for the appropriate Reactor Maintenance Department personnel, this issue remains an open item.

Reactor Training & Procedures Department (RTAP) has identified routine and special training needs for all Reactor Maintenance Department personnel. The maintenance training program is divided into three areas: administrative, core, and specialty. All maintenance personnel will be required to complete or test-out of the basic administrative and core training courses. Specialty training will be conducted to maintain a qualified, well-rounded maintenance organization. The continuing training needs of maintenance personnel in their support of the tasks assigned to the maintenance department is addressed by training outlined in these programs. RTAP has published two curriculum models for the Reactor Maintenance Department: one for Electrical & Instrumentation

(12) Maintenance Tools and Equipment Control

Maintenance tools and equipment control shall be developed and implemented for corrective actions deemed necessary as a result of the self-assessment findings.

(13) Station Material Condition Inspection

A plan and schedule shall be developed and implemented for the corrective actions deemed necessary as a result of the self-assessment of the station material condition.

The reactor housekeeping program restart criteria are described in Section 10.5.

(14) Management Involvement

WSRC shall develop and implement a program to increase management involvement and knowledge of plant status and evolutions. This program shall require management visits to the plant and specify: (1) minimum frequency and duration of visits, (2) process for reporting findings/observations made during the visits, (3) management levels required to participate in the visits, and (4) disposition and resolution process for findings/observations. These requirements shall be in accordance with the methodology specified in ROMP MS-13.1.

(15) Maintenance History

A plan and schedule shall be developed and implemented for the corrective actions deemed necessary as a result of the self-assessment of the reactor maintenance history program.

(16) Analyses of Maintenance Problems

A program shall be developed and implemented to provide for a systematic analysis to determine the root causes of unplanned occurrences related to maintenance. Root cause determinations shall be performed on equipment failures not having a clearly identifiable, and/or understood cause of failure. This program shall have provisions for information collection, information analysis, cause determination, corrective action, and corrective action follow-up.

A training program shall be developed and implemented to train specified reactor maintenance personnel in root cause determination.

13.3 Evaluation

Maintenance Department Organization and Administration

Many activities are being evaluated in support of the restart effort in the area of Maintenance Department Organization and Administration. These areas are required to implement an effective and credible maintenance program.

(7) Control of Maintenance Activities

A Control of Reactor Maintenance procedure shall be developed and implemented.

(8) Post-Maintenance Testing

A post-maintenance testing program shall be established that is applicable to all maintenance activities and addresses each department's responsibilities, equipment to be included, degree and type of testing, procedures, acceptance requirements, testing control, and results documentation.

(9) Procurement of Parts, Materials and Service

A plan and schedule shall be developed to implement the corrective actions deemed necessary as a result of the self-assessment of the procurement of parts, materials, and services. These corrective actions shall be in accordance with the WSRC Quality Assurance Manual, 1Q (Reference 6).

(10) Material, Receipt, Storage, Inspection, and Handling

The reactor maintenance organization shall establish procedural guidelines for proper control of materials that are to be used in the reactor areas. These procedures shall include attributes that ensure that: (1) materials that are in the custody of the reactor maintenance organization are stored and maintained in such a manner as not to degrade the material or its ability to perform its designed function, (2) reactor maintenance personnel perform inspections of the material prior to obtaining custody of the material and prior to installation of the material, and (3) the material is handled in such a manner to ensure it will not be degraded. This program shall conform to the requirements of WSRC Quality Assurance Manual, 1Q.

(11) Control and Calibration of Measuring and Test (M&TE) Equipment

A program shall be established to control measuring and test equipment when it is in the possession of the reactor maintenance department. The program shall also specify requirements for: (1) M&TE selection, (2) proper storage and control of M&TE equipment in the reactor areas, (3) functional checks of the equipment prior to use, and (4) requirements to prevent the contamination of the M&TE equipment. The program shall provide for actions taken to: (1) return M&TE equipment for calibration, (2) perform a timely assessment of M&TE equipment usage when M&TE equipment is found to be out-of-tolerance, and (3) control subcontractor-provided M&TEs. This program shall be in accordance with the WSRC Quality Assurance Manual, 1Q.

(2) Training and Qualification of Maintenance Personnel

Reactor maintenance personnel shall complete reactor maintenance training in accordance with the commitment in the ROMP item HP-8.1c.

Training shall be completed for those programs that are identified as being required for reactor restart as a result of the reactor maintenance self-assessment.

(3) Maintenance Facilities

Office space and equipment shall be evaluated, and the maintenance coordination groups shall be located in the reactor areas.

(4) Corrective, Predictive, and Preventive Maintenance

(a) The backlog of corrective and preventive maintenance shall be reviewed to determine which items must be completed prior to restart.

(b) WSRC shall develop a safety-related equipment list in accordance with the requirements of Section 5.6. This safety-related equipment list shall supersede those systems specified by the Item MS-6.1 Savannah River Plant K Reactor Restart Strategy (also known as the Charlotte Criteria). WSRC shall also develop a program for the use of a safety-related equipment list for reactor maintenance activities.

(c) The conduct of maintenance procedure resulting from the reactor maintenance self-assessment shall be developed and implemented.

(d) The preventive maintenance program shall be evaluated against INPO 85-032 (Reference 5), and a plan and schedule for implementation of any required corrective actions shall be developed.

(5) Maintenance Procedures

(a) Maintenance procedures used on safety-related equipment shall be reviewed, revised, or prepared to incorporate the following minimum requirements: (1) retest/return to service requirements, (2) control of measuring and test equipment (M&TE), and (3) Quality Assurance (QA) verification hold points.

(b) A plan and schedule shall be developed and implementation initiated for the corrective actions deemed necessary as a result of the self-assessment.

(6) Planning, Scheduling, and Coordinating Maintenance

Maintenance program and procedures for planning, scheduling and coordinating maintenance shall be developed and implemented. A maintenance coordination group shall be established in each reactor area.

13 MAINTENANCE

13.1 Background

Reactor maintenance has a primary role in ensuring safe and reliable plant operation. To carry out this role, a strong maintenance philosophy and culture (a set of standards, values, and convictions) based on sound and respected principles should be developed at all nuclear facilities. Achievement of excellence in maintenance requires a team effort and a dedicated commitment to raising standards.

Dupont committed to improving the reactor maintenance program at the Savannah River Site (SRS) based on the results of audits performed by the National Academy of Scientists and Engineers (NAS&E) and the Institute of Nuclear Power Operations (INPO). In February 1989, Dupont issued an SRS-wide Maintenance Improvement Plan identifying eight key elements that were in need of improvement. One of these elements was the performance of a self-assessment by the Reactor Maintenance Department based on the elements identified in INPO 85-038, Guidelines for the Conduct of Maintenance at Nuclear Power Stations (Reference 1). The self-assessment was to include a review of: (1) work control procedures, (2) maintenance requests, (3) supervision of maintenance activities, (4) review of completed maintenance requests, (5) temporary repairs, and (6) control of contractor personnel. This assessment (Reference 2) was completed in January 1990. The results of the assessment, which were compared against draft DOE Order 5480, established the foundation for the Reactor Maintenance Improvement Plan (MIP) (Reference 3), which was issued the end of April 1990.

13.2 Restart Criteria

The criteria for determining the adequacy of the SRS reactor maintenance program for restart are described below.

General

Prior to restart, a Maintenance Improvement Program shall be established that incorporates the elements specified below. The implementation of this program shall be initiated at restart for safety-related systems identified in Table 5.6.1. Reactor Operations Management Plan (ROMP) items addressed in paragraphs 13.2.2 and 13.2.5 below shall be satisfactorily implemented. Additionally, adequate control of maintenance activities and post-maintenance testing activities shall be in place.

(1) Maintenance Department Organization and Administration

Reactor maintenance organization goals, policies, and objectives shall be developed in accordance with INPO 85-001, "Performance Objectives and Criteria for Operating and Near-term Operating License Plants" (parts MA.1-10) (Reference 4).

12.4 Conclusion

WSRC has established programs in compliance with Restart Criteria 5, 8, 10, and 11. ROMP item SE-12 and WISRC-33 address Restart Criteria 1, 2, 3, 4, 6, 7 and 9. The proposed programs, procedures, or provisions established by WSRC, as described in the evaluations of the individual restart criteria, will satisfy the acceptance criteria. This SER subject will remain open pending review of the issued procedures as listed in the individual acceptance criteria sections.

12.5 References

1. WSRC, Reactor Engineering Procedure RP 5.1101, "Conduct of Reactor Engineering Activities," Rev. 0A (draft).
2. WSRC, Procedure RD 5.02, "Request for Engineering Assistance (REA)," Rev. 0A.
3. WSRC, Manual DPSP-82-1002, "Training and Operations Guide."
4. WSRC, "Reactor Operations Management Plan (ROMP)," Rev. 3, April 10, 1990.
5. WSRC, Memorandum OPS-RO-900067, from J. J. Gallagher to Distribution, "Upgrading of the Emergency Response Organization," April 19, 1990.

by procedure DPSP-82-1002, Section 1. If procedure RP 5.1xxx for the Equipment Performance Monitoring program is not implemented prior to restart, WSRC has compensatory provisions in place to satisfy the requirements of this Restart Criterion. Compensatory measures for satisfying this criterion before restart include the following:

- o Periodic inspections and tests for safety-related equipment required by the 105-1851 series DPSOLs.
- o Operations surveillances in accordance with Reactor Operations Standing Instruction (ROSI) II.8.
- o Scheduled Equipment Maintenance tracked by the Work Management System.
- o Technical Surveillance defined in Section V of the Training and Operations Guide, DPSP-82-1002.
- o Chemistry Performance Indicators are in place.
- o The Reactor Systems and Components Monitoring program is being upgraded to the ASME Inservice Inspection (ISI) program level. (This item is also related to Restart Criterion 3 of SER Section 4, "Inspection and Testing.")
- o Unusual events or conditions are reported and investigated under the Unusual Occurrence system in accordance with DOE Order 5000.3A. This reporting system includes the Reactor Events Reporting activities.

The above-mentioned items are the compensatory measures for satisfying this criterion before restart. The formal development of the reactor performance monitoring program is listed in the ROMP as open item 12.1.1-02.

Restart Criteria 8, 9, 10, and 11

WSRC has established an Engineering Assistance Request program. This program is guided by procedure RDP 5.02, "Request for Engineering Assistance." This procedure defines the responsibilities of the initiator(s), the REA coordinator, the responsible engineer(s), and their supervisor(s). The demarcation of these responsibilities and the methods of control and tracking of the REA activities prescribed in this procedure satisfy the requirements of Restart Criteria 8, 10, and 11; however, the procedure does not require periodic assessment by the Manager of RED as required by Restart Criterion 9. WSRC is expected to revise procedure RDP 5.02 to satisfy this requirement. The staff has reviewed the draft revision of this procedure. Also, the revision of this procedure is listed as an open item in ROMP 12.1.1-03. Criteria 8, 10, and 11 are closed; however, until the revised draft of RDP 5.02 is approved, Restart Criterion 9 remains an open item.

Restart Criterion 4

Restart criterion 4 requires WSRC to have a significant portion of the engineering and technical support personnel to be within its own organization, with supplemental skills provided as required from outside organizations. As of September 30, 1990, there were 453 people working in the Reactor Engineering Department (RED). As of March 1, 1990, there were 1297 SRL employees and 5728 E and PD employees (including include Bechtel contracted personnel). The staff has determined that about 1/4 of the employees in the above Divisions are providing supplemental skills from outside organizations. This level of staffing satisfies Restart Criterion 4.

Restart Criterion 5

Restart Criterion 5 requires WSRC to establish administrative controls to ensure effective utilization of the engineering and technical support by the operating group, including a list of experts to be used in the control rooms. Procedure RP 5.1101 provides interface methods for conducting engineering and technical support activities. The staff determined that the list of experts has been compiled and is in use.

A Reactor Engineering Technical Experts List Procedure is listed as a deliverable in the ROMP, 12.1.1-01. The staff has reviewed and commented on the draft of this procedure. The contents of this draft prescribe the methods for the establishment, use, and maintenance of the Technical Experts List. Since both procedures (RP 5.1101 and ROMP 12.1-01) are still in draft form, this is an open item, pending review of the final issued procedures.

Restart Criterion 6

Restart Criterion 6 requires WSRC to have an engineering and technical support group available to provide timely assistance and consultation. The group shall be of sufficient size and technical knowledge and shall be competent in all disciplines important to safe and reliable reactor operations. As discussed in the evaluation of Research Criterion 4, above, WSRC has sufficient expertise both within and outside RED to provide competent and timely support. Memoranda OPS-RTD-900117 and OPS-RES-900066 define duties of the area technical and area engineering support groups. Inter-office Memorandum OPS-RO-900067, dated April 19, 1990 (Reference 5) from J. J. Gallagher to Distribution indicates that the Emergency Response Organization (ERO) has been upgraded. This memorandum also defines the duties of the cognizant personnel for the Emergency Operations Facility and the Technical Support Center. The staff determined that these WSRC activities satisfy the requirements of Restart Criterion 6.

Restart Criterion 7

Restart Criterion 7 requires WSRC to establish a reactor performance monitoring program. The staff determined that RED will conduct the equipment monitoring program according to the procedure RP 5.1xxx. Until this procedure is in place, these monitoring tasks are controlled by the Manager, RED. A reactor event report monitoring program is in place. This program is guided

Restart Criterion 1

The staff's evaluation of the program elements addressing Restart Criterion 1 included a review of: (1) Chapter 13 of the Safety Analysis Report (SAR), "Conduct of Operations"; (2) WSRC Technical Specifications (TS), Category 5 (5.0, "Administrative Controls"); and the Reactor Engineering Department (RED) draft of procedure RP 5.1101, "Conduct of Reactor Engineering Activities."

The implementation and control of technical support activities are prescribed in TS 5.3.1.3, "Reactor Engineering"; TS 5.3.2, "Savannah River Laboratory (SRL)"; TS 5.3.3, "Engineering and Project Division (E&PD)"; and TS 5.3.4.2, "Safety Department."

The focal point of technical support is the Reactor Engineering Department. Procedure RP 5.1101 prescribes the method of conducting design, support, and surveillance activities. This procedure also prescribes the interfacing methods among Operations, RED, SRL, and E&PD.

On the basis of this review, the staff determined that the contents of WSRC documents on the implementation and control of technical support activities will satisfy the requirements of Restart Criterion 1 when draft procedure RP 5.1101 is finalized and issued. This is an open item.

Restart Criterion 2

Restart Criterion 2 requires WSRC to establish policies describing the expected standards of technical support activities. Technical Specification 5.0 states that (1) SRS activities are considered to be a system of integrated processes, people, facilities, and procedures; (2) administrative controls are achieved through vertical and horizontal communications; (3) safe operational and accident boundaries are carefully defined; and (4) principles of defense-in-depth and redundancy shall be applied. The staff determined that these policy statements satisfy the requirements of Restart Criterion 2. The staff found that the guidelines established by Manual DPSP-82-1002, "Training and Operations Guide" (Reference 3), prescribe personnel qualifications, training requirements, and job descriptions. These guidelines fulfill the above-mentioned policy statements.

Restart Criterion 3

Restart Criterion 3 requires WSRC to establish formal programs to implement and control technical support activities. The staff found that these programs have either been established or are in the planning stage. WSRC documents RP 5.1101, DPSP-82-1002, and RDP 5.02 describe these programs.

These items are listed in the ROMP(Reference 4), memo from P. Dickson to J. Gallagher, OPS-RRP-900156, 09/25/90. This is an open item until draft procedure RP 5.1101 is finalized and issued.

- (6) An engineering and technical support group shall be available to provide timely assistance and consultation. The group shall be of sufficient size and technical knowledge and competent in all disciplines important to safe and reliable reactor operation. The group shall assist in areas such as the following:
- o Surveillance Test Program
 - o In-Service Inspection Program
 - o Plant Modification Control
 - o Setpoint Control
 - o Technical Review
 - o Reactor Engineering
 - o Computer Software Control
 - o Root Cause Analyses
- (7) A reactor performance monitoring program shall be established to optimize overall reactor performance as well as system and individual component performance and reliability. Specific parameters which are to be monitored through this program shall be identified.
- (8) An engineering assistance request program shall be established to provide a means for reactor personnel to submit documented technical inquiries and design evaluation requests for technical information and evaluations of potential problems when insufficient technical information or expertise is available within the requesting group.
- (9) The reactor engineering manager shall periodically assess the timeliness of resolution and backlog of assistance requests and reassess priorities or obtain additional resources.
- (10) Responses to engineering assistance requests shall receive technical review and approval by the designated engineering manager prior to returning them to the originator.
- (11) Upon completion of all required actions the engineering assistance request shall receive closeout approval by the reactor engineering manager who shall ensure proper documentation and storage.

12.3 Evaluation

The staff determined that WSRC has established a new procedure to comply with the restart criteria. The staff reviewed the draft of procedure RP 5.1101, "Conduct of Reactor Engineering Activities" (Reference 1), which includes a punchlist. The procedure and punchlist specifically address Restart Criteria 1, 2, 3, 4, 5, 6, and 7. Restart Criteria 8, 9, 10, and 11 are captured by procedure RDP 5.02, which addresses the Request for Engineering Assistance (REA) program (Reference 2).

The results of the staff's programmatic review are discussed below.

12 ENGINEERING AND TECHNICAL SUPPORT

12.1 Background

The primary technical support organization at the Savannah River Site (SRS) is the Reactor Engineering Department, which is providing direct engineering and technical support to the restart effort. Two other organizations within Westinghouse Savannah River Company (WSRC), the Engineering and Projects Division and the Savannah River Laboratory (SRL), provide technical support to the Reactor Engineering Department. Historically, the quality and quantity of the technical support resources for SRS reactor operations has generally exceeded that found at commercial nuclear sites. However, sometimes coordination among the support groups and the operation organization has not been adequate to ensure timely response. Many times the compartmentalized organizational structure impeded the full and integrated application of technical resources. Because of the age of the SRS reactors and their various complex core arrangements, the DOE staff has conducted several safety reviews, and WSRC has provided more rigorous procedural guidance to facilitate safe operation and to address potential safety concerns (e.g., improved management of engineering and technical support).

12.2 Restart Criteria

The criteria for determining the adequacy of the SRS engineering and technical support program for restart are described below.

- (1) The organization and administration of technical support shall ensure that a high level of performance is achieved through effective implementation and control of technical support activities.
- (2) Policies shall describe the standards expected for reactor technical support activities.
- (3) WSRC shall establish and maintain effective implementation and control of technical support activities through written guidelines, trained personnel, periodic monitoring and assessment of performance, accountability of personnel for their performance, and other resources as required.
- (4) A significant portion of the engineering and technical support personnel shall be within the operating contractor's organization with supplemental skills provided as required from outside organizations through established agreements in place.
- (5) Appropriate administrative controls shall be established to ensure the effective utilization of the engineering and technical support group by the operating organization. This program shall include a list of technical experts in the control room indicating the name, area of expertise, and means to contact available experts to supplement the technical support on shift. This list shall be available in the Central Control room. This list shall be verified and updated monthly.

Figure 1. Power and Flow During a Large Break LOCA.

Figure 2. F1 OSV Criteria Experimental Data Base.

Figure 3. ECS Wall Temperature Experimental Data Base.

Figure 4. Relation Between FI Analysis Codes.

Figure 5. Comparison of Measured and Calculated Plenum Pressure Distribution.

Figure 6. Relative Contributions of Various Assumptions and Uncertainties on the Power Limit.

TABLE 4

Sensitivity of Reactor Power Level to Minimum Assembly
Flow and Confidence Level

<u>Postincident Minimum Assembly Flow, gpm</u>	<u>Best Estimate</u>	<u>One Sigma</u>	<u>Two Sigma</u>	<u>Three Sigma</u>
15	97%	57%	43%	29%
20	132%	77%	59%	40%
30	191%	112%	85%	58%
40	255%	149%	113%	78%
50	312%	183%	139%	95%

- 100 percent reactor power implies an annual averaged power of 2400 MW. Based on expected radial power shapes this corresponds to a MARK 22 preaccident assembly power of 6.3 MW.
- All sensitivity case calculations were made based on plenum inlet break with A/C pumps on.

TABLE 3

Application of Standard Engineering Correlations
to LBLOCA Code Calculations

<u>Category</u>	<u>Accident Phase</u>	<u>Application</u>	<u>Correlation</u>
HEAT TRANSFER CORRELATIONS	FI	single phase forced convection	Sieder-Tate
		highly subcooled boiling	Bergles & Rohsenow interpolation technique
		fully developed boiling	Mikie & Rohsenow
	ECS	forced convection - churned turbulent regime	modified Dittus-Boelter
ON SET CRITERIA CORRELATIONS	FI	onset of nucleate boiling	Bjorge & Rohsenow
		onset of significant voids	SRL experimental ST=0.00455
	ECS	churned turbulent regime	SRL experimental determined regime map
FRICTIONAL LOSS CORRELATION	FI	isothermal frictional loss	Moody
		heated wall effect	Dormer
	ECS	two phase effects	Moody with phasic multiplier

TABLE 2
Relative Power Limits for Major DBAs

<u>Accident</u>	<u>Relative Limiting (OSV) Based Power Level</u>	<u>Core-wide Confidence Level to Damage (One Assembly or Less)</u>
DEGB LBLOCA (FI)	1.00 (by definition	.84
(ECS)	1.20	To be determined
LOPA	1.35	To be determined
PSB	1.36	.95
GRW	1.51	.999

Table 1

SUMMARY FOR CURRENT REACTOR CONFIGURATION
[50% (1200 MW) PRE-INCIDENT REACTOR POWER LEVEL]

Component	Lower Estimated (°C)	Best Estimated (°C)	Upper Estimated (°C)	Melting Temp (°C)
Mk 22 Target/Fuel	50	80	100	600
Mk 22 USH	280	360	450	600
Mk 60B	380	470	580	600
Septifoil Housing	550	650	780	600
Control Rod	720	850	1000	600
Safety Rod Sheath	670	770	830	1400
Thimble	460	580	680	600
Instrumenta- tion	*	*	*	*
Tank Wall	430	510	570	1400

* To be determined.

10. DPSTY-100, "Systems Analysis, Control and Safety Rod Cooling for the Savannah River Plant Reactors," October 1974.
11. WSRC-RP-90-1292, "SRS Position on the Implementation of the Single Failure Criterion," November 27, 1990.
12. Letter from Frank R. McCoy to Paul Rice (WSRC), "Single Failure Criterion," February 11, 1991.
13. DPST-87-737, TRAC Calculations of the Initial Flow Decay in L 1.1, December 1987.
14. WSRC-TM-90-5 (DPSTM-160), "A Probabilistic Method for Determining Effluent Temperature Limits for Flow Instability in SRS Reactors" (U), (draft), February 1990.
15. "Savannah River Restart Physics Review Rev. 2, LOCA FI Limits Methodology," September 12, 1990.
16. LANL Group N-12, "Independent Reactor Physics Analyses in Support of Savannah River Restart," April 1990.
17. DPST-85-828, "Revised Decay Heat Tables for Mark 16B, Mark 22, and Mark 31 Assemblies for Use in ECS Analyses," August 1985.
18. DPSOL 105-MC-2-LK, Reactor Incident, Revision 22, January 9, 1990, and DPSOL 105-MC-4-PLK, Major Process Water Leak, Revision 13, January 10, 1990.
19. WSRC-RP-89-796, "Analyses of the Pump Shaft Break and Gang Rod Withdrawal Incidents with TRAC-FLOWTRAN" (U), September 1989.
20. WSRC-TM-90-4, "LOCA FI Limits Methodology" (U) (draft), February 1990.
21. DPSTM-140, "FLOWTRAN: An Algorithm for Describing the Thermal-Hydraulic Behavior of SRP Assemblies" (U), September 1989.
22. WSRC-RP-90-XXX, "Preliminary Benchmarking of FLOWTRAN with MARK 22 Mockup Flow Excursion Test Data from B&W," to be issued.
23. EGG-EAST-8556, "Benchmarking the Six-Loop, Three-Dimensional RELAP 5 L-Reactor Model with Savannah River Reactor Test Data," August 1989.
24. Fuel-Coolant Interaction Tutorial, January 24-25, 1991.

Expected Safety Margin

For the FI and ECS phases, a large safety margin is imbedded in the conservative bounds of coolant flows and local powers and the limiting single failures, which are not likely to occur. Presently, there is no basis for assessing the safety margin associated with the LOPA and (preliminary) gamma heating limits.

11.3.4 Conclusion

On the basis of the staff's review, the power limits issue is determined to be open pending satisfactory resolution of the open items identified above.

The staff will continue to review the progress of efforts in the power limits area and report on the results of that review in the final SER or in a supplement to the SER. Resolution of the open items identified above will resolve the power limits issue.

11.3.5 References

1. WSRC-RP-89-373, "Action Plan to Upgrade T/H Limits for SRS Production Reactors," June 1989.
2. National Academy of Science/National Association of Engineers, "Safety Issues at the Defense Production Reactors," 1987.
3. WSRC-RP-89-1287, "Flow Instability Analysis of a Design Basis LOCA for the K-14.1 Subcycle" (U), December 1989.
4. Letter from J. L. Gallagher to R. E. Tiller, dated November 30, 1989, subject "Reactor Restart Power Level."
5. NES-ECS-890196, "K-Reactor Restart Initial Power Limits: FLOWTRAN-TF Assembly Calculations for the ECS Phase of a DEGB-LOCA," December 1989.
6. Memo from F. Witmer to J. P. Knight, "Concerns Over Gamma Heating Issue and Need for Office of Engineering and Operations Support (DP-12) Technical Review/Involvement," January 23, 1991.
7. DPST-88-666, "FLOWTRAN Benchmarking with Onset of Flow Instability Data from 1963 Columbia University Experiment," October 1988.
8. DPST-88-886, "Comparison of the FLOWTRAN Predictions of the Onset of Significant Voiding (OSV) to Savannah River Heat Transfer Laboratory Subcooled Boiling Flow Instability Measurements, Part 1" (U), Rev. October 1988.
9. WSRC-RP-89-870, "FLOWTRAN Benchmarking with the Onset of Flow Instability Data from 1988 Columbia University Single-Tube OFI Experiments" (draft).

responsive to reviewer comments (both WSRC internal and external) to produce a hierarchy of safety documents that are in agreement with accepted practice. The adequacy of the documentation remains an open item.

Confirmation of Codes and Limit Values (Restart Criterion 6)

Code verification and validation efforts would normally be expected to uncover any gross coding or modeling errors. Because of the lead time required to complete such an effort, it has been proposed that, in the interim, confidence in the codes is established through a comparison with separate, confirmatory calculations performed by LANL, which constitutes an independent group of analysts. Gross errors are expected to be identified and resolved through this method. This process should provide reasonable assurance that any significant errors have been detected.

The validation of the accuracy of the calculated power limit values resulting from the application of the reference codes suite is to be accomplished through having a third party, LANL, calculate power limits for the same thermal hydraulics boundary conditions. LANL will use the same safety criteria and a different set of codes/models, which has substantive differences from those adopted by WSRC. If reasonable agreement between calculations is achieved, the independent effort will establish a basis for confidence in the results. If there is significant disagreement, then both sets of codes will require an introspective review/evaluation to understand the differences and arrive at a consensus on the preferred technical approach.

Discrepancies between code-set power levels for a very complicated problem like the LBLOCA are not anticipated to be excessive considering both code families are benchmarked to the same set of conditions and the accident transient is not significantly different from the benchmarking. In the case of the FI phase, the rapidity of the transient and timing remain critical. With the ECS phase, two-phase flow and air-water separations phenomena in the loops, plenum and tank muff are a potential area of concern for direct application of the codes. For gamma heating, the estimate and treatment of tank air flows is critical along with realistically assessing radiative interchanges between components. The comparison to separate confirmatory calculations has not yet been completed. Pending completion of this effort and resolution of any concerns identified as a result of this effort, this is considered to be an open item.

Long-Term Cooling Phase and Accident Management (Restart Criterion 8)

Restart Criterion 8 requires a long-term evaluation of the consequence of a LBLOCA. This has not been done and is an open item. Of particular concern are single failures associated with primary pump and sump pump reliability resulting from a LBLOCA with the plant in a mitigative configuration, under the assumption that this could result in the limiting condition for the LBLOCA. The long-term management of potentially radioactive waste water effluents is also a concern for both the LBLOCA and LOPA accidents and is an open item.

Gamma Heating Power Limits Uncertainties

While WSRC has not yet discussed their criteria for "success," calculational methodology, boundary conditions, and assumptions inherent in their gamma heating scoping calculation studies, there are a number of potential pitfalls in the analysis that DOE staff and supporting consultants will be evaluating during the impending review process. The controlling heat transfer mechanisms; namely, convective and radiative exchanges are difficult to model in the complex geometry and variable boundary condition within the moderator tank. These mechanisms ultimately will require at least some limited experimental verification. Large variabilities of 50 to 100 percent from analytical projections are not unreasonable.

Depending on the conservatism represented in the modeling and computational analysis and the estimation, quantification, and/or bounding of uncertainties integral to and/or subsequent to the analysis, it may be necessary to apply an appropriate safety margin to the gamma heating power limits to fully account for uncertainties. As stated in the "Controlling Safety Criteria" section of this evaluation, the conservative accounting for uncertainties associated with gamma heating power estimates is an open item.

Documentation (Restart Criterion 9)

While the major portion of the analysis development effort for the setting of FI power limits is complete, the progressive hierarchy of documents that gives defensible methods, followed by models, benchmarks, core analysis, and limits is not fully available at this time. Much of the information provided in this section is based on information obtained during presentations and from draft documents, which are not acceptable documentation for concluding restart criteria are satisfied. This concern also applies to the current ECS and LOPA documentation.

Documentation for the gamma heating analysis does not exist.

The basis for the OSV safety margin, set at 1 sigma or 84 percent on a core-wide basis for a single assembly going into FI has not been formally documented.

For the ECS power limit, many of the draft ECS documents require revision to reflect the evolution of the methodology. This is especially true for those describing the treatment of uncertainties.

For the LOPA, current documentation is judged incomplete and is far from producing a defensible method, model, benchmark, safety analysis, and core limit. While the LOPA method, model, and benchmarking can draw heavily on FI-related technology, the safety analysis and resultant core limit must fully reflect system interaction and human factor vulnerabilities.

These observations are consistent with very early concerns over the completeness, quality, and timeliness of the safety documentation for the setting of the T/H power limits. Special emphasis needs to be given to being

uncertainties (i.e., whether the distributions are skewed and in particular, the nature of the "tails"). WSRC has performed a series of Monte Carlo calculations, which established that the overall FI uncertainty distribution can be approximated by a normal (Gaussian) distribution and that the nature of the "tail" can be appropriately quantified. Consequently, when one decreases power (ΔT_{as} for a given flow) by a bias to achieve a confidence level that the critical effluent temperature will not be exceeded, the probability of fuel damage associated with the actual high-side tail can be established.

Under certain conditions during FI, cavitation can occur. As a result of cavitation, the pressure boundary conditions for the assembly model are a function of power. Because a cavitation model is not included in the system code and because it is impractical to iterate between the system code and the assembly code, a code called UNCERT Version 16.3 was developed. UNCERT Version 16.3 uses power-dependent boundary conditions to mimic the effects of cavitation to incorporate the effects of cavitation into the analysis. The effects of cavitation on the boundary conditions are determined from a simple model.

ECS Power Limits Uncertainties

As in the case for the FI phase, a critical factor in determining ECS power limits is an accounting of unknowns and safety margins. Appropriate safety margins should be introduced as a cumulative power reduction from nominal subchannel power limits to statistically account for "hot" subchannel factors (e.g., fuel loading, channel eccentricity, subchannel coolant flow, and heat transfer), model omissions, and uncertainties from model/experiment benchmarking. Also, for the ECS phase, special attention must be given to the change in D₂O coolant properties due to injection of ECS polybor solution and to ECS water quality considerations. The analysis and treatment of uncertainties and their effect on ECS power limits is not nearly as developed for this latter phase of the accident as it is for the FI phase. At present, the ECS power limit is projected to be higher than the gamma heating values. If the margin between the gamma heating and ECS limits are 3-sigma or greater, the probability of fuel damage during the ECS phase will be acceptably small for the anticipated restart power levels. While Table 4 implies this is the case, it still needs to be verified by qualitative means that power levels dictated by minimum (worst) assembly flow values at high confidence levels also will produce a low probability of fuel failure on a core-wide basis. This is an open item.

For the ECS phase, the random variables that apply to a given assembly are combined statistically by linear error propagations (a conventional square root of the sum of the squares formula) to determine an overall uncertainty allowance. Inherent in this analysis is the assumption that covariance is negligible and variation from the reference case is linear. The listing of uncertainties for the ECS phase is not nearly as exhaustive as the listing for FI. The ECS uncertainty analysis needs to show that it is bounding with respect to "linear" assumptions and that the inclusion of critical assumptions is complete. This is an open item.

$$T_{cr} = E_{Tin} + T_{in} + B_{3D} B_{hw} B_{void} \Delta T_{as}(T_{in}, x_F, x_{cav})$$

where

T_{cr} = critical effluent temperature

E_{Tin} = error in the effluent temperature measurement

T_{in} = assembly inlet temperature

B_{3D} = 3-D multiplier

B_{hw} = heated wall multiplier

B_{void} = void effect multiplier

ΔT_{as} = temperature rise for an axisymmetric assembly

x_F = parameters in the axisymmetric model

and

x_{cav} = cavitation parameters

An uncertainty analysis was performed on the simplified assembly model represented by the equation. The uncertainties were categorized as:

- o manufacturing uncertainties, including tolerances, geometric alignment, and material composition
- o data base uncertainties, including solid properties, fluid properties, heat transfer correlations, and flow parameters
- o uncertainties in input obtained from other codes, including TRAC-PF1/MOD1 and neutronics codes
- o uncertainties resulting from model idealization, including nodding, and convergence criteria
- o errors in process measurements used to set the limits

A list of 89 uncertainties was originally evaluated and considered in the setting of FI power limits. Several additional uncertainties have been recently included in the analysis. Most notable of these are an assembly effluent temperature basis and an inlet plenum temperature bias, which reduce FI power levels by 0.75 percent and 2.25 percent, respectively. As one would expect, most of the uncertainties have little effect on the FI power limit. Figure 6 illustrates the relative effect of the more important uncertainties. For restart, the uncertainty in fuel cylinder loading variation will be decreased from ± 15 percent, as used in the analysis, to ± 12 percent consistent with the actual fuel loading. This will reduce the B_{3D} multiplier.

A key to estimating the probability of fuel damage from the FI statistical treatment is the actual statistical distributions of the particular

If propagation of melting can be terminated without the occurrence of a liquid metal-water interaction, any fuel assembly damage that may occur will be well within the capability of the airborne activity confinement system (AACS) to contain radioactive particulates. Reactor power, coolant inlet and outlet temperatures, flow rate, and fuel burnup are the controlling variables that limit the potential fuel/core damage associated with the family of DBEs. By controlling reactor power and coolant outlet temperature, the potential for fuel/target/component damage is reduced to an acceptable level.

At this juncture, no discussion can be provided on the evolving modeling methodology being used by WSRC. Both the independent review committee and DOE staff and consultants will be reviewing this effort in detail. LANL is in the process of performing an independent confirmatory analysis, adapting commercial thermal hydraulic codes wherever possible. The gamma heating methodology remains a open item.

Uncertainties (Restart Criterion 7)

A crucial element in the setting of a safe power level is a statistical accounting of uncertainties and biases inherent in the plant and fuel configuration, analysis assumptions, and code solutions. To be conservative and ensure a high probability of not exceeding controlling safety criteria, the calculated power limit is significantly reduced, e.g., for FI, base power levels are reduced approximately 40 percent to account for nonidealities and uncertainties. The evolving state of affairs for the gamma heating limits makes it difficult to make specific statements on how uncertainties are being assessed and accounted for.

FI Power Limits Uncertainties

TRAC and FLOWTRAN-FI are used to generate a best estimate of the effluent temperature for an assembly. This calculated temperature is adjusted for uncertainties to produce a conservative operating limit. These uncertainties arise from manufacturing tolerances, experimental errors, approximations, model omissions, process measurement errors, etc. A statistical analysis method is used to quantify the uncertainties in the effluent temperature (Reference 14). The uncertainty in the effluent temperature is used to ensure that the probability of not exceeding the damage criteria is 84 percent on a whole-core basis.

From a realistic point of view, it is not practical to routinely run three-dimensional FLOWTRAN-FI calculations to establish power limits because of the large amounts of computer time involved. Instead, two-dimensional FLOWTRAN-FI calculations are performed, and the results are adjusted by a multiplication factor to arrive at the three-dimensional condition. The multiplication factor was determined by comparing detailed three-dimensional calculations under a variety of conditions to the corresponding two-dimensional calculations. Similarly, corrective multiplicative factors have been developed for heated wall effects and void effects. As a result, the critical effluent temperature is given by:

Gamma Heating Methodology

Because of the initiation of this effort by WSRC in early July with notification to DOE in late September 1990 and the intensive on-going activity ever since, WSRC has not provided DOE with technical documentation of the gamma heating analyses. Only the most cursory of reporting of findings has been provided.

Ultimately, the methodology must provide assurance that gamma heating of core components that are not directly cooled during the ECS phase of a LOCA will not disrupt ECS coolant flow to the fuel and targets, create disruptive forces through metal water reactions or hydrogen explosives that could eject control or safety rods or otherwise damage the core, and produce conditions leading to a severe accident.

Recent simplified scoping analyses have indicated that the safety rods and thimbles, septifoils, Mark 60B target assemblies (thermal shield), and instrumentation thimbles are at risk of overheating (see Table 1).

Relatively high temperatures are projected for the tank wall and top plate, which could present difficulty with rapid tank refill and resultant steam production. The tank draindown and subsequent gamma heating phase for the LBLOCA occur with the displacement of the heavy water by light water through ECS addition. With the light water and the tank drained down, there is no concern for recriticality. (For an intermediate LOCA, this situation probably does not apply as the tank can drain down without activation of the ECS.)

Recent and continuing work by Sandia National Laboratories for the Office of New Production Reactors and for SRL indicates that for 6 kg quantities of molten aluminum, larger, explosive, chemical-driven interactions are possible with water. Energy releases of approximating 30 percent of the potential chemical energy along with hydrogen production have been observed (Reference 24). At this juncture, there is speculation that gaseous impurities (i.e. fission products, elements with high vapor pressures, such as lithium, magnesium, etc.) will disrupt the passive oxide film and create fresh surfaces of nascent aluminum to sensitize the tendency to initiate (trigger) and sustain such a reaction. Preliminary analyses indicate that fuel melting can lead to bridging across the fuel channel to block coolant flow and lead to melting of a target. With steam binding of the assembly, collapse of the steam void could bring water into the melting fuel/target channel. It is not necessary for the molten aluminum to fall to the bottom of the assembly. Hydrogen, which is a byproduct of the reaction, poses an additional uncertainty and concern. A fuel assembly typically contains 44 kg of aluminum. It remains to be demonstrated that if small quantities of aluminum-based components melt during gamma heating phase they will do so in a benign way and not create a mini-explosion that could initiate a severe accident. This is an open item. The possible sensitivity of molten aluminum-water interactions to impurities and gas producers may be significantly lessened because the fuel and targets are directly cooled, and the weak control rods (aluminum-lithium alloy) can be driven from the core as a mitigative measure.

involved, they do a reasonably good job of approximating the system configuration and permit the codes to be correlated and benchmarked to the real system.

At present, once "tuned" to the experimental data, RELAP5 has been able to accurately predict coolant flows and system pressure drops at different moderator tank levels under two-phase flow conditions. The recent finding of the pump outlet break as being the worst ECS break location resulted from a recent retuning of the code to more specialized 1989 L-Reactor flow test data. However, there has consistently been difficulty in predicting air flow rates accurately. RELAP5 is predicting air flow rates several times those observed. This is suspected to be due to phase separation in the pipe network and/or pumps that are not properly modeled. Under conditions of low flow to the plenum, the in-reactor flow test data indicate that stratification occurs in the plenum. In this instance, liquid flow to the assemblies is dominated by gravity weir flow through the assembly upper-end fitting. Therefore, modeling air flow rates accurately with RELAP5 is less of a concern. At low tank levels, the foot of the assembly is no longer trapped, and air flow is facilitated out the assembly into the tank region further complicating the model. At this point in time, a confirmatory modeling effort by LANL appears to be homing-in on matching the air flows estimated for the tests.

At high flow rates, flow in the plenum changes to dispersed flow. Under these conditions, liquid flow rates are sufficiently large that cooling the core during the ECS phase of a LOCA is not a problem. Although, ideally, it would be desirable to accurately model both the air and water flow, limitations in both the test data and analytical models hinders such an effort. While test data have shown that the discrepancy in air flow is not an important factor for the ECS phase, it remains necessary to document a conservative minimum assembly coolant flow rate. As indicated previously, air flow is an important consideration for the gamma heating phase. For the ECS phase this is an open item requiring documentation.

A summary of the correlations applicable to the FI and ECS phases is presented in Table 3 to provide an appreciation for the technical bases, namely, the application of standard codes and correlations with appropriate benchmarking.

Because the methodologies for both the FI and ECS phases are still under development and represent "state-of-the-art" thermal hydraulic analyses, DOE has convened an independent review panel consisting of five renowned experts in the thermal hydraulic and reactor safety areas to provide a peer review of the analysis methods and their experimental data base. A report on the panel review of the FI portion of the LOCA was completed in June, 1990. In general, the panel has accepted the FI methodology as a valid means to set power levels. Review of the ECS methodology and experiments was completed in August 1990. The panel did not endorse the WSRC recommended ECS power limit of 59 percent, but rather recommended a power limit of 50 percent based on an engineering judgement after a review of all available OTE data and a consideration of uncertainties. Resolution of the outstanding recommendations and comments from the independent peer review panel is a confirmatory item.

Once the boundary conditions are fixed (i.e., local decay heat generation rate, inlet coolant temperature, channel coolant flow rate, plenum pressure, assembly pressure drop and inlet air void fraction), FLOWTRAN-TF iterates between cells/nodes to obtain an axial temperature distribution for the fuel assembly subchannel surfaces as a function of time. In the calculation for power limits, the assembly power is progressively increased in increments until one of the subchannel surface nodes equals or exceeds the ECS T/H criterion (i. e., the fuel surface temperature exceeds the coolant saturation/boiling temperature for the static pressure conditions at that axial location ($T_{wall} \geq T_{sat}$)).

The pressure-drop correlations used in the assembly code are based on conventional single-phase, friction-drop correlations and form losses from the literature multiplied by standard two-phase flow multipliers. Similarly for heat transfer, a conventional single-phase literature correlation has been applied with an experimentally based correction factor using phasic liquid velocities and a standard hydraulic diameter. In the churn-turbulent flow regime, the empirical multiplier of the heat transfer correlation is increased based on test data from prototypical annular channels heated on one side. A correlation was obtained with maximum variations of the order of plus 56 percent or minus 36 percent. The observed "churned-turbulent" heat transfer coefficient is 15 percent higher than one would estimate for single-phase forced convection. This is not unreasonable considering the additional gross turbulence induced from two-phase mixing and rib cross flows. One must consider the current correlation as a strictly empirical approach that works for the limited test data base. This correlation does not take into account double-heated and/or asymmetric heated channel wall effects. Literature values suggest these effects may be expected to decrease the effective single-phase heat transfer coefficient by 10-15 percent. The heat transfer coefficients to be used in the ECS phase that use phase-based velocities to correlate the existing data base and the questions regarding single and double wall heating are open items that need additional development and require documentation.

The effect of variable oxide film on the fuel surface is expected to produce nonuniform heating of the coolant in the subchannel, which may have a secondary effect on the heat transfer coefficient, but more importantly will create significant local surface temperature variations. It is not apparent that the effects of irregular oxide film formation and flaking have been fully considered in the ECS limit-setting process. This has been addressed in technical reviews in the FI phase power limit case and the hydraulic effect of tighter channels has been recently accounted for in the uncertainty analysis through a 1.5 percent power reduction. The insulating effect of an aluminum oxide film is estimated to account for a power reduction of 0.5 percent, which is judged to be counterbalanced by overconservatism inherent in the analysis. For the ECS phase this remains an open item.

As previously described, the RELAP5 and modified RELAP5 models and codes set the boundary conditions that are critical to the FLOWTRAN-TF assembly model. These codes have been subjected to an extensive benchmarking effort based on full-scale primary system hydraulic tests. While these tests do not fully simulate the transient nature of the LBLOCA accident, and no heating is

the limiting fuel assembly. As previously indicated, loss of a DC primary pump as a single failure further exacerbates the flow maldistribution, as would the inoperability of the ATS to two ECS valves on separate ECS addition lines, if indeed credible. This issue is being addressed as part of the single failure criteria issue.

The prediction of the onset of thermal excursions is based on local conditions in the subchannels. Coolant flow from the upper plenum through the assemblies is in a gravity-dominated regime. The falling coolant "streams" that get "captured" by individual fuel-target channels are quite sensitive to several variables, i.e., total assembly liquid flow, total assembly air flow, static pressure at foot of the assembly, and plenum pressure.

It is expected that local power and coolant inlet temperature have only a secondary effect on initial capture distributions. Flow variations between subchannels of the same channel at the top of an assembly, while quite severe initially (± 30 percent), have been observed in separate effects tests to equilibrate several feet down along the channel as a result of rib cross flow. The formation of steam associated with OTE in a single subchannel affects coolant flow in neighboring subchannels (both radially and azimuthally). Thus, as previously indicated, the entire assembly is assumed to experience OTE if OTE occurs in a single subchannel. It has not been firmly established that a partial isolation of subchannels due to in-reactor bowing of assemblies does not increase the tendency for bubble stagnation and OTE. This concern is not germane to the $T_{\text{wall}} = T_{\text{sat}}$ criterion, other than evaluating the safety margin inherent in the $T_{\text{wall}} = T_{\text{sat}}$ criterion.

In estimating limiting flows to a subchannel for in-assembly calculations, RELAP5 is used to determine the net loop coolant flow to the plenum as a function of moderator tank-level coupled to a modified RELAP5 nodal model of the upper plenum coolant level (Reference 23). The upper end fitting of the fuel assembly, which consists of a vertical three-slot sleeve and a multiple orificed cylinder in series, is resolved into a single slot gravity weir, which correlates reasonably well with plenum coolant level and assembly flows (independent of air flows).

WSRC has developed a special code, FLOWTRAN-TF, based on the conservation of mass, energy, and momentum to account for two-phase flow, heat transfer effects, and cross-rib gap flows in assembly subchannels. The heat conduction models developed for FLOWTRAN-FI have been incorporated in FLOWTRAN-TF. Each subchannel coolant node has radially adjacent fuel surface temperature nodes to accommodate the heat transfer in the cell. Rib fin effects are also handled in the same manner as they are in FLOWTRAN-FI. In order to initiate the computation, an air void fraction must be assigned to the computational cell above the fuel. This is done by assuming an air void volume in the first (top) axial cell as adjusted by an experimentally determined partitioning factor. Results from FLOWTRAN-TF have been shown to be relatively insensitive to the value assigned. Two-phase flow across the ribs is modeled by the application of an assumed partition factor based on values given in the literature.

accommodated by resolving the rib, based on fin efficiencies, into an equivalent additional surface of the fuel cylinder to which the rib is attached. FLOWTRAN-FI has the capability to model in three dimensions, the radial, azimuthal, and axial heat production and conduction in the fuel. For thermal hydraulic effects, equally-spaced coolant temperature nodes are positioned radially adjacent to fuel surfaces to accommodate a heat "split" between adjacent channels. Conventional hydraulic diameter, frictional pressure drop, and heat transfer correlations are used for nonboiling water flowing in annular channels. The onset of subcooled nucleate boiling, which precedes the occurrence of OSV, is accommodated by smoothly transitioning from the forced convection correlation to a literature correlation for fully developed boiling. Standard frictional loss correlations are used for nonheated and heated walls. Because of the high flow rates and pressure drops, no consideration is given to flow across the ribs between subchannels during the FI phase of the accident. This is implicit in the use of full channels in the analysis and application of the OSV criteria. FLOWTRAN-FI is not capable of handling conditions under which steam voids form.

Based on test data from single tubes and single heated wall annular channels (References 7 through 9), a Stanton Number value of 0.00455 has been selected as a bound for predicting OSV. This value was conservatively selected as 30 percent lower than the best estimate value for this data base. Subsequent testing of representative flows has not produced an FI observation that is less than the 0.00455 value. It should be appreciated that the Saha-Zuber correlation is simply a bound by no exception. A mechanistic correlation would be expected to collapse the data base. Statistical analysis of SRS operating data base indicates that a large number of cases of FI should have been observed in the reactors while at steady state power during the 100 plus reactor years of operation at SRS. No case of FI in a geometrically nominal fuel assembly has been observed. This indicates an inherent, yet to be quantified, margin in the OSV/FI criteria/methodology as applied to heavy water systems.

Tests at Babcock and Wilcox (B&W) (Reference 22) simulated the approximate flow decay and power transient in a single full-size, electrically-heated fuel assembly mockup. The FIs were approximately 10 percent higher than those predicted by FLOWTRAN-FI using OSV as a predictor of FI. While the coolant flow recovered after about 0.1 seconds in the B&W tests, due to system hydraulic "bounce-back," there was generally no recovery from FI once it occurred. The heater cylinders in the B&W tests were Monel, as opposed to an aluminum alloy used for the SRS fuel cylinders. The good conductivity of the SRS fuel might possibly provide some margin to recovery from FI.

ECS Phase Methodology

The ECS phase analysis is based on a series of calculations of the primary system tank and piping with air ingress caused by the inability of the ECS to fully make up the coolant lost out broken pipe. Air enters the system through the vents and plenum inlet line and markedly degrades system loop flow. Because of the asymmetric flow distribution due to flow out the break, a trough develops in the plenum liquid level adjacent to the inlet region of the broken pipe. This significantly reduces the quantity of coolant getting to

steady-state. A bounding approach has generally been taken in estimating the transient powers.

The actual accident transient and system response dynamics cannot be evaluated in a full-scale test. Instead, the various elements that comprise the system are characterized independently to develop a "bounding best estimate" analytical model that can fully describe the time-dependent behavior of the system. The strength of the overall model depends on the individual models having a firm physical and mechanistic basis to permit meaningful extrapolations to accident conditions. When empirical correlations are used, the data base must be representative of the conditions expected during the transient or bound the conditions, and scaling parameters, if applicable, must be delineated and fully understood. Alternatively, one can demonstrate a decoupling of effects experimentally and/or analytically. Examples of decoupling components are the USH assembly inlet and the tank outlet "muff" that under stratified ECS flow conditions behave as weirs.

The methodology used for the gamma heating phase of the accident is still evolving.

FI Phase Methodology

The FI phase methodology is illustrated in Figure 4 and described in detail in Reference 20. The methodology uses the TRAC-PF1/MOD1 code to describe the physical conditions in the piping system and flow distribution in the plenum and the FLOWTRAN-FI code to model conditions in the assemblies. An asymmetric pressure and flow distribution develops in the plenum as a result of the pressure drop across the forest-like interstructure in the plenum and the azimuthal variation caused by the "alignment" of the fuel assembly inlet housings to the plenum inlet nozzles. TRAC has been accurately correlated to full-scale, in-reactor fluid flow tests (Figure 5) and is used to predict the asymmetric flow distributions in the plenum. The apparently large scatter in the data of Figure 5 at the periphery of the core illustrates the effect caused by the azimuthal alignment of the coolant inlet nozzles with the fuel assemblies. TRAC calculates a mean pressure value for each the 30 ring-sector nodes comprising the plenum "face." These values establish boundary conditions for the FLOWTRAN-FI calculations.

Assemblies are classified into groupings called flow zones for evaluating conditions in the fuel assemblies to the T/H criterion, rather than treating each assembly's flow and power individually. The lowest assembly flow rate for a specific flow zone and the highest power is conservatively used to characterize and bound all assemblies in that zone for power limit calculational purposes. Once the limiting assembly and its power are identified, assembly effluent temperature limits are established for all assemblies in the flow zone. The total core power is calculated based on the limiting assemblies in each flow zone.

FLOWTRAN-FI (Reference 21), a WSRC-developed code, calculates the assembly thermal response and coolant hydraulics. The code divides the fuel assembly into concentric fuel cylinders spaced apart by parallel coolant channels. The presence of integral rib spacers (fins) on different fuel cylinders is

It should be noted that the PSB was selected because it represents the most severe incident normally analyzed for transient protection analysis.

Other DBEs, such as the loss of septifoil cooling, will be reanalyzed integral to the SAR Chapter 15 upgrade process using improved calculational procedures and codes. It is not anticipated that limits from these DBEs will eclipse the gamma heating DEGB-LBLOCA power limit, which currently dominates restart.

Acceptability of Accident Analysis Methodology (Restart Criteria 3 and 6)

While this discussion focuses on the FI DEGB-LBLOCA analysis, because of the formative nature of the gamma heating and ECS analyses, the single-phase conditions in the primary system make the resultant methodologies and models developed for the FI phase of the LOCA applicable to the analyses of other DBEs. More specifically, as previously indicated, this technology has been applied to the LOPA, PSB, and GRW accidents.

As previously discussed, the approach taken to analyze the LBLOCA FI and ECS accident scenarios has been a bounding best estimate approach with uncertainties. As one reviews and understands the technical bases and assumptions implicit in these analyses, the inherent conservatism in the resultant power limits become apparent. Because of the differences in the controlling physical phenomena for the FI and ECS phases of the LBLOCA accident, somewhat different methodologies have evolved in modeling the two phases of the accident, which can almost be considered independent sequential accidents. For both phases, a similar approach was taken in developing the methodology. The methodology includes the following:

- o Using existing industry and NRC accepted computer codes to model the overall T/H behavior in the reactor vessel and primary piping.
- o Modifying these codes when necessary to approximate inlet plenum assembly pressure/flow distributions.
- o Applying experimentally derived flow distributions and loss coefficients to the tank moderator region and outlet muffs.
- o Developing specific models and codes to predict the overall and local coolant flow and heat transfer conditions within the fuel assembly.
- o Applying experimentally based T/H criteria to local conditions in the assembly to predict the maximum acceptable power limit.

The development and tailoring of these models and codes to the accident has been based on both system tests and separate effects experiments.

Neutronic heat input data, which are crucial to the analysis, come from current physics and properties codes. Because of the excellent core-wide instrumentation at SRS (each fuel assembly has four outlet thermocouples and flow indication), the radial power distribution is finely tuned operationally, and integrated axial powers are known on an assembly-by-assembly basis for

The rationale justifying a smaller decay heat uncertainty for both the FI and ECS phases rather than using the recommended additional 20 percent margin of Appendix K has been developed based on industry revisions to decay heat standards as well as on data, but this has not been documented. This is an open item.

Sensitivity of AC Pump Trip Timing (Restart Criterion 5)

Currently, procedures (Reference 18) call for the PW alternating current (AC) pump drive motors to be tripped manually early in the accident (within the first 2 minutes). This reduces the venting of air into the primary Process Water (PW) system and the subsequent degradation of the PW pumps, thereby allowing a higher initial core power for the ECS phase of the LBLOCA. Also, additional safety margin for the more probable small break LOCAs is provided with this procedure. The direct current (DC) motors, which also drive the primary coolant pump, remain on. The flow provided under either AC and DC or DC motor drive is essentially equivalent, as the flow is primarily limited by tank exit restrictions, not pump drive capacity. However, secondary effects have been observed in the 1989 L-Reactor tests in that the DC pump flow rate readjusts on the loss of the AC drive. Consequently, the PW pump speed with only DC motors needs to be determined to be consistent with analyses. This is an open item. Preliminary results based on an independent system studies by LANL are that the minimum assembly flow is sensitive to the timing of the AC pump trip. Additional studies by LANL and WSRC are expected to further clarify this consideration. The appropriate timing of the AC Pumps in the event of an accident is an open item.

Enveloping Nature of the Accident (Restart Criterion 2)

Analysis of the LOPA, pump shaft break (PSB), and gang rod withdrawal (GRW) accidents (Reference 19) DBAs, which traditionally defined the SRS reactors' maximum power limit envelope, were analyzed using similar T/H criteria, modeling, and computational techniques (TRAC, FLOWTRAN-FI and physics code suites) as those developed for the LBLOCA FI phase analysis. Preliminary results are shown in Table 1.

The calculated LOPA limit is based on the installation and operation of automatic 190 and AC pump trips. Operator activation of the ECS system is scheduled 60 seconds after receiving the requisite alarm which signals a sump water level of 25 inches. These limits are contingent on high reliability of the sump level indicator/alarm system and the assumption for the operator response time which includes limited diagnostics. Because of these vulnerabilities, no core-wide confidence level can be assigned until the appropriate sensitivity studies reflecting the Manual Incident Action (MIA) and system uncertainties are completed. It should be noted that the relative power level given in Table 2 for the LOPA is based on the assumption that the same uncertainties as evaluated for the LBLOCA prevail. Even based on the uncertainty of credible single failure considerations, it is unlikely that the uncertainties for the LOPA could be larger than those associated with the gamma heating phase of the DEGB-LBLOCA, and a more detailed or bounding assessment is probably not required. However, the lack of documentation regarding uncertainties makes these concerns open items.

region. It also needs to be confirmed that a break in the PW loop containing an ECS inlet represents the worst case and that specific combinations of broken loop and operating ECS lines do not combine to produce adverse flow distributions in the plenum. Determination of the worst break location is an open item.

Local Power Levels (FI) (Restart Criteria 4.a and 4.b)

During the FI phase, the power level within an assembly is a function of (1) time into the accident scenario and (2) initial axial and radial power distributions and local effects due to control rods, etc.

Of critical importance in establishing conservative power limits is the initial power distribution within an assembly. Conservative upper bound delay scram times, as prescribed in the technical specifications, are used to maximize the power-flow mismatch. Non-ideal behavior and "hot" spot considerations (e.g., subchannel fuel hot-stripes and eccentricities, assembly azimuthal power tilts, etc.) are fully accounted for in the treatment of uncertainties, which significantly reduces the power from the nominal bounding best estimates case (Reference 14).

For the FI calculation, the power limits are not overly sensitive to decay heat values, but the decrease in fission heat generation rate as the safety rods enter the core is critically dependent on the scram circuit timing. A bounding best estimate approach with a full accounting for uncertainties has been taken. There is justification not to include the additional 20 percent margin in the decay heat generation rate as recommended by Appendix K (Reference 4). Los Alamos National Laboratory (LANL) has reviewed the WSRC physics calculations to confirm the power generation distributions in the assemblies and the general kinetic behavior of the SRS reactors (Reference 15). A detailed kinetic behavior confirmation, including evaluation of reactivity temperature coefficients, is to be completed by LANL prior to restart; however, local power distributions will not be confirmed until after restart (Reference 16). Because of the conservatism of this analysis, this is considered to be satisfactory for restart.

For the ECS phase, timing is critical from the state of the thermal hydraulics (i.e., coolant and air flow to the plenum), but the heat generation rate is past the initial drop caused by the insertion of the safety rods and is no longer highly sensitive to time. By using calculations methods prescribed in American Nuclear Standards Institute/American Nuclear Society (ANSI/ANS) standard 5.1, WSRC is able to predict decay heating effects as a function of fuel composition and fission product content (Reference 17). Axial, radial, and azimuthal decay heat generation distributions are consistent with worst case allowable preaccident distributions. An independent review by the Idaho National Engineering Laboratory (INEL) has confirmed that gamma heating effects, which can be appreciable, have been correctly considered in calculating the decay heat distributions in the fuel and targets. However, additional analysis is needed to address the non-fuel, non-target components. This is an open item.

base from which to identify what serious consequence active component failures are probable and what their frequency of occurrence is on demand.

WSRC has stated that for the FI phase, the failure of a single safety rod to drop is the prevalent single failure. Combined with technical specification limitations and configuration considerations, this single rod failure could result in a total of nine rods in a cluster not dropping. For the ECS phase, WSRC used the failure of an ECS valve to open as the limiting credible single failure. Each of the four ECS addition lines have two valves in series (a gate valve and a ball valve). Valve failure on demand has typically been 2.5 percent, which translates into a 90 percent chance for all three ECS addition lines to operate after the break. In the WSRC initial FI and ECS analysis, it is assumed that the break occurs in a line containing one of the ECS addition lines, leaving three ECS lines available for the injection of ECS coolant. In addition, one of these three ECS lines is assumed to fail to operate per single failure criteria considerations.

WSRC is in the process of systematically identifying single failures that can affect the outcome of the DBEs (Reference 11). These single failures are being screened to determine if any are more limiting than the failures selected in the reference analyses. For the ECS phase, the loss of a DC primary pump now appears more limiting than the loss of an ECS addition line in that the larger flow reduction to the plenum aggravates the trough in the plenum level which under stratified flow conditions dominates minimum assembly flow. The extent to which these single failures will be fully incorporated in the analyses for limiting power levels is uncertain.

DOE has provided WSRC guidance for the application of single failure criteria (Reference 12). For DBEs that are likely to occur, single failure vulnerabilities must be corrected. For those DBEs that are not likely to occur, a Justification for Continued Operation (JCO) must be submitted for DOE approval along with a corrective action plan. Thus, the technical basis demonstrating that the single failures selected by WSRC for each phase of the LBLOCA are credible and limiting needs to be documented. This is an open item and is a requirement of Section 11.1 of this SER.

It is not apparent that credible single failure criteria have been considered in determining LOPA, PSB and GRW power limits. It needs to be confirmed that single failure criteria have been identified and given due consideration. This is an open item.

Worst Break Location (Restart Criterion 4.c)

The worst break location for the FI phase appears to be on a plenum inlet line. This has been determined as a result of a series of sensitivity studies (Reference 13). Recent calculations for the ECS phase indicate that the pump outlet break is probably the worst location; however, this might change with continued benchmarking and refinement to the system hydraulic code (RELAP5) based on continuing analysis of the 1989 L-Reactor tests. This particular break location appears to exhibit extreme sensitivity to air in-leakage to the plenum through the piston seals which depresses the plenum coolant level distribution at the center and markedly reduces assembly coolant flow in this

accident analysis restart criteria discussed in Section 11.1 of this report. The adequacy and the reliability of the sump level instrumentation has not yet been satisfactorily demonstrated and is an open item.

The possibility that motor control centers and other equipment within the basement of the reactor building could possibly flood and short-out, thereby causing a mini-blackout that incapacitates ECS before it is initiated by operator action has not yet been addressed and is considered an open item.

The LOPA, which also relies on ECS injection for cooling, shares common concerns with the LOCA related to long term cooling. In fact, for a large inlet header break in the secondary coolant water system, ECS pumps A and B flood out within 3-5 hours, depending on operator response. After this time, coolant is provided to the core by the ECS booster pump, the 190 pumps, and/or the river/Par Pond distribution system.

DEGB-LOCA (Restart Criterion 4.c)

For the SRS reactors, which operate in a subcooled condition at low temperature and pressure, the probability of a double-ended, guillotine break (DEGB) is equal to and most likely less than that for the recirculation line in a commercial boiling-water reactor (BWR) (i.e., 10^{-4} per reactor full power year). There is rationale to define initiating conditions (break opening time and break size) that are less severe than those applied to power reactor DEGB LBLOCA. Best estimates place the probability for a piping DEGB-LOCA at SRS in the range of 10^{-6} to 10^{-7} . A further, although not easily quantifiable, conservatism for the probability for a DEGB-LOCA at SRS under restart conditions, i.e., 50 percent power, is that the primary system will be operated at a maximum of 150-160 degrees F as opposed to 195-205 degrees F for full power. For these lower temperatures, the primary system qualifies as a moderate energy system which historically has not been required to accommodate DEGB considerations.

However, because a fully acceptable technical basis supporting leak before break has not been developed, the K-reactor restart power limits will be set based on the historic, instantaneous DEGB LOCA. While it is recognized that the low pressure SRS reactors pay a significant penalty in the FI and ECS power limits by adopting a nonrealistic break, as long as gamma heating concerns dominate the setting of the power limits, break size and break dynamics play a significantly lesser role.

Single Failure Criteria (Restart Criteria 5)

In developing the FI and ECS LBLOCA scenarios, limiting credible single failure criteria were considered as opposed to worst possible failures. The SRS plants were designed and built 35 years ago. The safety rod system possesses some potential for common mode failures of scram activation/enhancement relays and common electrical bus vulnerabilities. The ECS system possesses an automatic transfer switch that activates valves on two separate ECS addition lines; thus, its failure could render two ECS lines inoperative. The long operational history at SRS provides a credible data

and the determination of the effectiveness of two-phase coolant flow to the septifoils, if coolant flow out the top of the septifoil housings is considered as a viable mitigative measure), many critical parameters must be characterized experimentally. In addition, if limited degradation and/or melting of select in-core components is contemplated as an acceptable risk, the basis for that determination, including the evaluation of the potential for liquid metal-steam/water interaction, needs to be evaluated and accepted by DOE. The technical concerns for the gamma heating phase are considered to be open items.

Long Term Cooling Phase

At this point in time, WSRC has not formalized long term cooling requirements or plans for the LBLOCA, which relies on once-through ECS cooling or makeup to keep the core and reactor tank internals from over heating. The ability to disassemble the core after the accident is not likely for an extended period of time. As a consequence, ECS water will overflow from the reactor and retention basin after about 3 days. Based on LBLOCA considerations for no fuel or target damage, radioactive releases will be low.

Loss of Pumping Accident (LOPA)

In a LOPA, the flow of process water through the core is lost. The Level 1 Probabilistic Risk Assessment (PRA) has identified a LOPA, initiated by rupture of a large cooling water (CW) system pipe that floods the process water pumps, as a significant contributor to the core melt frequency. Because of this, the mitigation and accommodation of this accident have assumed greater importance. Analysis of the LOPA tentatively predicts a higher limiting power level than that associated with the LBLOCA. The analytical modeling tools and codes developed for the LBLOCA analyses have been applied in assessing preferred response options to the initiating CW LOCA.

Recent analysis of the LOPA indicates that there is an "ECS phase" that may result in very low local assembly flows that possess the potential for flow reversal. The limiting OSV criteria previously considered applicable and controlling is being re-evaluated and may have to be replaced with a set of thermal excursion criteria (including flooding limits) similar to those used for the ECS phase of the LOCA. This problem can significantly reduce LOPA power limits and is being considered integral to the resolution of the open issues that accompany the LOPA.

The CW system has been modified to initiate an automatic trip of the 190 secondary coolant pumps and a load shed (trip) of AC primary pumps on detection of a large line break in the CW system. A level indicator alarm will be installed in the control room to indicate that a 25-inch flooding level has been achieved in the sumps to require the operator to initiate ECS actuation before the DC primary pumps shortout because the flow decay associated with DC pump trip under flooded conditions is too abrupt to permit the manual ECS activation to overcome the loss of DC flow. This operator action must occur within 60 seconds of the alarm indication to ensure the core is cooled. This execution occurs 6-8 minutes into the accident, which is less than the 10-minute period requiring special review under the transient and

conditions being potentially "hotter" than the "nominal" and for uncertainties inherent in the model and analysis (Restart Criterion 4.d).

In the ECS phase, the controlling mechanism for the onset of thermal excursion (OTE) appears to be a local stagnation of gravity-dominated flow due to the creation of stationary steam-air voids at an axial location in a subchannel. The ECS T/H criterion, which limits the fuel cladding surface temperature to values below the coolant saturation temperature, prevents steam generation and OTE (Figure 3). For restart power levels, this criterion also appears to be more restrictive than the flooding and CHF criteria that are also imposed (Restart Criterion 4.d).

It remains to be documented that there are no heat flux dependent local surface phenomena that can initiate OTE. ECS separate effects experiments have thus far been performed in channels with a single heater wall at heat fluxes twice the values expected in the core for the ECS phase of a LBLOCA initiated from 50 percent power (Restart Criterion 4.d). This is an open item.

Critical heat flux (CHF), a long-exercised, experimentally based, T/H criterion specific to SRS, is also satisfied for both the FI and ECS phases. At power levels of 50 percent or less, there remains ample margin (Restart Criterion 4.d).

Gamma Heating Phase

The strongest source of gamma heat is attributed to radioisotopes with half lives of a few hours, at most; as a consequence, reactor power level is a primary parameter in the level of subsequent gamma power. The abundance of fission products, which track burnup, is a secondary contributor to the production of gammas in the short term. It is, however, the major source of gamma heating during the long term cooling phase.

When the reactor moderator tank drains, both the cooling and shielding effect of the moderator are eliminated, and exposed components begin to progressively heatup through the absorption of gamma radiation until natural convection, conduction, and radiation heat transfer to thermal "acceptors" or sinks within the tank achieve a steady state condition. For several components melting and/or component failure may occur prior to thermal equilibrium. This phenomena was identified as a concern during an earlier study on safety and control rod cooling requirements (Reference 10).

WSRC is currently attempting to better understand the gamma heating phase, the resolution of which is currently indeterminate. Gamma heating concerns also apply to certain intermediate small break LOCAs in which the moderator tank can inadvertently partially drain.

In the gamma heating phase, there are a number of technical concerns that need to be addressed as indicated below and the reference analysis for power limitations must be documented. Due to the complexities and uncertainties associated with this problem (e. g., convection with limited net air flow; radiation between near white bodies with low and time variant emissivities;

response to additional information derived from ongoing modeling, benchmarking and experimental efforts. This is considered an open item.

The worst break location for the ECS phase has been only recently studied in necessary detail. For a LOCA at a pump discharge, the leakage of air from the moderator tank to the plenum via the piston seals appears to suppress the coolant level in the center region of the plenum and introduce an inherently large uncertainty. It is this coolant level that is a critical variable in dictating limiting assembly coolant flows.

For the LBLOCA, the acceptable power level limit that has been proposed is that which results in a reasonable probability (84 percent or better on a core-wide basis) that no fuel assembly is damaged. This probability applies to the FI phase of the LOCA. This probability has not been determined for the ECS phase. This is an open item.

Currently, it appears the FI power limit will be higher than the ECS power limit, which in turn will be higher than the gamma heating power limit, which will probably dictate the restart power. The contribution to the probability of fuel damage during the FI and ECS phases (for a DEGB LOCA starting from the gamma heating power limit) is expected to be negligible. The probability of fuel damage during the controlling gamma heating phase can not be estimated at this time because of the complexity of the physics and the preliminary nature of the analysis. Sufficient engineering safety margin needs to be incorporated in the establishment of the power level to ensure that, in the event of the LBLOCA, fuel and target damage (melting) is unlikely. The establishment of an appropriate safety margin for the gamma heating phase is an open item.

The actual initiation of fuel damage is considered on a local subchannel basis for both the FI and ECS phases. The assumption is made that once gross steam generation begins in a subchannel, it adversely affects coolant down-flow within the channel, and ultimately leads to coolant flow reduction to the entire fuel assembly. It is conservatively assumed that a disruption to subchannel coolant flow induced by gross steaming is paramount to fuel assembly damage with the possible exception of flow reversal during the ECS phase of the LOPA. Gross steaming produces two different effects for the FI and ECS phases: (1) marked two-phase pressure drop increases for FI and (2) flow stagnation, typically referred to as "flooding," from gas buoyancy effects for ECS.

In both cases, the net effect is the same: a stifling of coolant flow to subchannels and ultimately to the assembly. Precursor criteria have been selected for the FI and ECS phases that preclude gross fuel and target surface dryout and any subsequent fuel damage.

For the FI phase, a conservative, test-confirmed value (References 6 through 9) of the Stanton Number is used to estimate the onset of significant voids (OSV) in a nominal fuel channel in downflow (Figure 2). The OSV criteria, which represents a criteria by non-exception, is invariably a precursor to, and predictive indicator of, an impending flow instability in the channel under nominal conditions. Power levels/limits associated with the nominal channel are subsequently reduced to statistically account for subchannel

assemblies. The FI phase is a precursor to the ECS phase; thus, the hydraulic conditions in the FI phase define the conditions in the early stages of the ECS phase. Coolant flow within the subchannels of the assemblies must be maintained to preclude local fuel temperatures exceeding a value above which the fuel cladding will not readily rewet. This is commonly referred to as the Liedenfrost temperature. The Liedenfrost temperature for aluminum is in the range of 200-240° C. Considerable forgiveness is provided by the high thermal conductivity of the aluminum alloy fuel and the low specific decay power of the fuel, which permits local "dry" patches to exist for several seconds before rewetting.

The hydraulics can be separated into flow in the "external loops" (which are comprised of the tank, piping, pumps, heat exchangers, ECS, and inlet plenum) and flow within the fuel assemblies (including upper and lower end fittings). The liquid flow in the "external loops" is amenable to modeling with appropriately benchmarked existing thermal hydraulic codes, e.g., RELAP and TRAC. Two-dimensional flow in the plenum and tank regions requires special adaptations of these codes to determine the time-dependent flow rate to each individual assembly in the core. Knowing the assembly inlet coolant flow rate, pressure, and temperature as bounding conditions permits the modeling of the thermal hydraulics within the assemblies. An engineering approach, based on empirical separate-effects data for channel coolant flow splits and onset of thermal excursion (OTE), has been deployed in the models. A detailed mechanistic assembly model, FLOWTRAN-TF, is being developed and will be benchmarked against the results of the ongoing test program prior to power operations.

Within the assembly subchannels, the gravity-dominated, churn-turbulent coolant and air flows are sensitive to perturbations caused by air heatup and coolant cross flow to adjacent subchannels through the variable (0-40 mils) rib gaps. As a consequence, it is difficult to quantify local heat transfer conditions. A semi-empirical approach, based on separate effects (single heated wall) experiments, is being taken to establish the T/H criteria for the ECS phase. For restart, a conservative T/H precursor criterion to potential fuel damage conditions for the ECS phase power limit have been adopted. The criterion is as follows:

Fuel surface temperature (T_{wall}) remains equal to or less than the local saturation temperature of the coolant (T_{sat}).

This criterion is more stringent than criteria that limit the effect of local and general flooding as well as critical heat flux effects in the disruption of assembly cooling.

The analysis of the ECS phase of the accident is based on a combination of bounding best estimate for loop flows and a mechanistic modeling and engineering approach for phenomena within the fuel assembly. Because of the nature of the uncertainties and the lack of prototype experimental data to perform a detailed statistical analysis comparable to that performed for FI, ECS uncertainties have been classified as biases and random uncertainties and treated in a conventional manner. Thus, the methodology is still evolving in

Flow Instability (FI) Phase

During normal operation, the single phase D₂O coolant flows downward at relatively high coolant velocities (e.g., 30 ft/sec). As the flow decreases as a result of the LOCA, the coolant temperature rises. If the coolant temperature reaches the local saturation temperature, steam voids can be produced. Void production further reduces the coolant flow rate, and a cycle of decreasing flow rates and increasing void production may lead to overheating in the channel. This process is typically referred to as Ledinegg flow instability. Because the flow rates remain relatively high during this phase and the coolant remains predominately in the single phase regime (i.e., subcooled liquid), flow and heat transfer correlations are relatively well known for the flow loops, plenum, and fuel assemblies. Also, the coolant flow distribution remains relatively unchanged except in the sector of the core near the break. A bounding, best-estimate, statistically-based accounting of all uncertainties associated with the analysis can be made based on a knowledge of actual fuel assembly geometry and loading variabilities, local neutronic effects, and loop hydraulic uncertainties.

Implicit with the implementation of a bounding best-estimate approach is the acceptance of very limited, probabilistic core damage. The state of the knowledge is commensurate with a best-estimate modeling of the accident and treatment of uncertainties, although some parameters are conservatively bounded.

A significant consideration is the thermal-hydraulic criterion for limiting power for the FI phase of the accident. Initially, a "no bulk boiling" criterion was assumed to apply to the individual annular coolant channels within the fuel assemblies. Locally, subchannels could produce steam prior to the channel mixing cup temperature reaching the saturation temperature. Later, correlations for the onset of nucleate boiling (ONB) were used as the T/H criterion to predict the onset of flow instability (OFI) within the subchannels. For restart, a prototypic, experimentally-confirmed, bounding limit for the onset of significant voids (OSV) has been adopted as a precursor to FI.

As the reactor system depressurizes immediately following the break, there is a possibility for cavitation to occur in high velocity regions downstream of the fuel where the subcooling is insufficient to preclude boiling. Steam voids that form at the pressure plates could conceivably be carried to the shell holes at the foot of the assembly and result in reducing coolant flow from the assembly. The presence of dissolved helium blanket gas, coming out of solution, could hinder any recondensation of steam voids. The potential for flashing in the assemblies and cavitation in the pumps and loop piping are also addressed in the FI limit setting process.

Emergency Cooling System (ECS) Phase

In contrast to the FI phase, the ECS phase of the accident deals with much larger system variabilities associated with two-component, two-phase flow. In the ECS phase, recycled primary coolant and air along with ECS injected water are pumped into the inlet plenum of the reactor and flow down through the

worst location: pump outlet break,
limiting single failure: DC pump break,
limiting power: to be determined

gamma heating limiting power: around 21%, with no
damage to fuel/target/components and no
modifications to core. This power limiting may rise
to 30% or higher if the septifoil housings are
modified.

LOPA limiting power: 62%

loss of septifoil cooling limiting power: 30% or higher
with modified septifoil housings

As evident from the above discussion, the power limits methodology is in an evolving state. As such, much of the information provided in this section of the SER is based on preliminary information, information presented at meetings, and information from draft reports. This has resulted in many issues remaining open, primarily because of the lack of approved documentation.

Controlling Safety Criteria (Restart Criteria 1 and 4.d)

The development of the power limits safety criteria is a two-step process: (1) the extent of acceptable core damage, a macro consideration, must be defined; and (2) the specific thermal hydraulic criteria that will conservatively bound the accident envelope within this acceptable core damage criteria must be established.

Large Break Loss of Coolant Accident (LBLOCA)

Historically, for the SRS reactors, a DBA core damage fraction of 1 percent has been associated with the LBLOCA, which is consistent with the capability of the airborne activity confinement system.

There are major physical phenomenological and timing differences between the FI and ECS phases of the LBLOCA, which carry over into the modeling and analytical treatment of the transient. Phenomenologically, the gamma heating phase is similar to the ECS phase in that two-phase flow at low moderator tank levels is controlling; however, the capability to dissipate gamma heating when the tank is drained is sensitive to the net air flow in the tank. Air flow in the tank has been only a secondary consideration in evaluating the fuel assembly temperatures during the ECS phase. Any of these three phases may be controlling in setting the LOCA power limit. Except, perhaps, for some environmental constraints, sufficient over-capacity for cooling exists during the longer term cooling phase, so this phase is not considered power limiting. A "no bulk boiling" T/H criterion and safety margin were initially applied to the FI and ECS phases of the accident.

meet tritium production rate requirements so that operation of the plant is practical, as opposed to keeping the plant down in order to make appropriate modifications to increase capacity. It is expected that this power level limit will increase as the capability to predict the SRS reactors' behavior under various accidents improves. Over the longer term, judicious modifications to the plant and operating procedures are anticipated to lead to a progressive restoration of power level.

Currently, there are three DBAs that appear enveloping and could potentially limit reactor power: the large break loss-of-coolant accident (LBLOCA) and the loss of pumping accident (LOPA), which are judged to very low probability events; and the loss of septifoil cooling accident, which has occurred twice during the life of the SRS reactors.

The loss of septifoil cooling accident, which has recently been identified as a limiting DBA because of the application of an inappropriate boiling heat transfer correlation, is included under SER Section 11.1, which addresses the family of non-LOCA accidents. The potential consequences of this accident are still being evaluated.

On November 30, 1989, based on preliminary calculations of power level (References 3 through 5), WSRC recommended power limits of 53 percent and 61 percent full core power for the FI and ECS phases of the LBLOCA, respectively. While there has been no written confirmation or documentation provided, recent estimates of acceptable power limits have varied widely and none of these "power limits" have been subjected to the scrutiny of a stringent external technical review and/or independent confirmatory calculations, as were the 1989 tentative FI and ECS power limits.

As of January 29, 1991, the following preliminary "controlling" power limits, based on a historic full power of 2400 MW representing 100 percent power, have been quoted by WSRC (it should be noted that the "actual" power limits generally vary with river water temperature).

LBLOCA

FI worst location: plenum inlet break
 limiting single failure: loss of ECS addition line
 limiting power: around 50%

ECS location: plenum inlet break
 single failure: loss of ECS addition line
 limiting power: 59%

 worst location: pump outlet break
 single failure: loss of ECS addition line
 limiting power: less than 50%

 location: plenum inlet break,
 limiting single failure: DC pump failure,
 limiting power: about 45%

- (d) The thermal hydraulic power limit shall provide a low probability of exceeding the fuel and target damage criteria. Fuel and target damage criteria shall be defined by physical parameters established in laboratory tests. Alternatively, mechanistic limits may be defined if demonstrated to be precursors to damage. A low probability of exceeding the damage criteria shall be defined and justified consistent with Section 11.1, "Transient and Accident Analysis."
- (5) The capability of operator and system mitigative response to the design bases accident shall be ascertained to be within that assumed in the analysis based on a best estimate scenario for the accident, both with and without the worst single active component failure. WSRC shall ensure that all power-limits related systems will function in a predicted and uniform manner, (i.e., one system will not interfere with the function of another).
- (6) The physical models and correlations used in the accident analysis should be based on prototypical tests. In lieu of such tests, limited prototypical tests, in conjunction with separate effects experiments and literature data, shall be used to develop and benchmark the computational model of the accident used to set the power limits consistent with Section 5.8.
- (7) Uncertainties shall be considered when setting operational power limits. The calculated best estimate or evaluation model calculated power level shall be adjusted to include appropriate allowances for basic calculational or model omissions, as well as analytical uncertainties.
- (8) The evaluation of design basis events (DBEs) shall include sufficient time beyond the event initiation to assess the potential for more restrictive conditions with respect to power limit determination.
- (9) The power limit settings resulting from the design basis accident analysis shall be documented.

11.3.3 Evaluation

The safety philosophy adopted for the SRS restart power limits is to operate at steady-state reactor power levels that will result in no significant core damage on a statistical basis, in the very unlikely event of a design basis event (DBE).

Determining a technically defensible maximum power limit is a crucial issue in restarting the reactors. This power limit represents an acceptably safe power level at which the reactor can operate. It must also be high enough to satisfy sensitivity requirements for critical core instrumentation and to

11.3.2 Restart Criteria

The criteria for determining the adequacy of the SRS power limits program for restart are described below.

- (1) The power level limits will conservatively preclude damage to fuel and targets during the course of the design basis event. The power level limits shall not exceed the limit assumed in the transient and accident analyses as discussed in Section 11.1, "Transient and Accident Analysis."
- (2) The enveloping nature of the design basis event relative to other events will be confirmed. The design basis event shall be a double-ended guillotine break (DEGB) loss-of-coolant accident (LOCA), unless an alternative basis is demonstrated to be applicable.
- (3) Methodology for establishing safe power limits shall be developed and documented as a part of the Thermal Hydraulic Power Limits Action Plan.
- (4) The approach recommended in 10 CFR 50, Appendix K, or more recently the best estimate plus uncertainties approach, shall be used in establishing power limits. Either an Evaluation Model or Best Estimate Model analysis will be acceptable. However, whichever is used, the methodology, detailed assumptions, correlations, uncertainties, margins, model omissions, etc. shall be documented and justified as providing conservative limits.
 - (a) If the 20 percent decay heat generic safety margin implicit in 10 CFR 50, Appendix K is not used, an acceptable safety margin shall be established using SRS reactor-specific margins and conservatisms based on measured decay heat generation rates.
 - (b) Time dependent heat sources and heat distributions shall be conservatively determined considering:
 - o initial stored heat
 - o fission heat reduction through control of reactivity
 - o fission and activation products and actinide decay
 - o heat input from targets, nonfuel hardware within the reactor, and frictional effects in the process cooling system
 - (c) Worst break locations and sizes shall be used. Conservative minimal break opening times shall be used for the dynamics of the full offset break.

11.3 Power Limits

11.3.1 Background

The power limits constitute a set of bounding steady-state power levels such that in the event of a design basis accident (DBA), the radiological safety criteria of Title 10 of the Code of Federal Regulations (CFR), Part 100 will not be exceeded. DBAs, while considered highly unlikely, are generally judged to be credible with probabilities of occurrence of greater than one chance in a million per reactor year.

The design basis events (DBEs) considered in establishing the DBA-power limits represent the full group of accidents that comprise the transient and accident analysis section (Chapter 15) of the SAR. Historically, in the operation of the SRS reactors, a number of power limits that originate from the different DBAs have controlled the maximum power level at which operation was permitted. Typically, the DBAs that have been considered are the loss of coolant accident, the loss of pumping accident, the gang rod withdrawal accident, and the pump shaft break accident.

Similar to commercial reactors, the plant design and engineered safety features provide high assurance that the core will remain in a coolable geometry after a DBA. Whereas commercial reactors have massive containment systems that control fission product releases from accidents orders of magnitude more severe than DBAs, the SRS reactors rely on an active confinement and a filtered vent system to control the release of airborne radionuclides during the course of any accident in which fuel is damaged. The SRS confinement system must be capable of removing sufficient airborne radionuclides to fully comply with 10 CFR 100 dose guidance at the site boundary for all DBAs. The fuel, targets and in-core internals of the SRS reactors are constructed of aluminum-based alloys, which have the potential to interact energetically and explosively with water/steam when the alloy has melted. To avoid the remote possibility of fuel or target melting, which creates a scenario for a bridge to a core melt accident that could possibly exceed the capacity of the confinement system, various conservative thermal-hydraulic (T/H) criteria have been established to preserve fuel and target integrity during all DBAs. The specific DBA that produces the lowest power limit consistent with the thermal-hydraulic criteria establishes the maximum power level at which the reactors can operate.

A major technical effort, consistent with commercial practice, has been made for the past 3-1/2 years to understand the complex phenomena associated with a large pipe break and the corresponding effects on the emergency core cooling. The analytical models and computer codes that have been developed and are being benchmarked to simulate the LBLOCA transient are also being adopted to project the bounds of the LOPA scenario and other lesser accidents. The technical scope of the code/modeling and experimental efforts are described in the Thermal Hydraulics Power Limits Action Plan (Reference 1). The Action Plan was developed when SRS personnel and the NAS/NAE review committee recognized the potential for steam binding in the assemblies during a LOCA (Reference 2). In March 1987, the reactor power levels were reduced to 50 percent pending further evaluation.

11.2.5 References

1. Letter, J.D. Spencer to C.E. Ahlfeld, "New Technical Specifications - Retire Technical Standards," February 3, 1988.
2. Letter, J.S. Moore to R.E. Tiller, "Technical Specification Improvement Plan," August 25, 1989.
3. Letter, P.W. Dickson to R.E. Tiller, "Technical Specification Interim Program Plan," January 15, 1990.
4. Letter, F. McCoy to P. Rice, "SRS Administrative Controls Technical Specifications Revision 3," November 16, 1990.
5. "Technical Specifications WSRC-TS-10003, Revision 0," December 18, 1990.
6. Letter, S. Richardson to P. Rice, "Technical Specification WSRC-TS-10003, Revision 0," January 25, 1991.
7. Letter, P.W. Dickson to F.R. McCoy, "Technical Specification Change Request," March 11, 1991.
8. Letter, S. Richardson to P.D. Rice, "K-Reactor Technical Specifications WSRC-TS-10003 Revision 1," March 15, 1991.

The use of the MERITS Program documents was previously endorsed for use by WSRC in the ROMP.

The current completion date for the third-phase of the TSIP is January 1992. The TSIP states that the post-restart program will include the following:

- (a) all Technical Specifications required by the NRC criteria
- (b) the development of bases for the surviving technical specifications
- (c) a review for completeness

During the course of the DOE review of Revision 0, DOE identified the need to consider additional Technical Specifications to cover specific operability requirements. The NRC's Interim Policy Statement shall be applied to these systems to ensure that sufficient Technical Specifications are included in the Phase 3 scope. The operability of these systems should be currently covered by procedures; therefore, Technical Specifications for the following are not required prior to restart:

- o Water Removal and Storage System (including sump pumps)
- o CW Recirculation pumps and diesel generators
- o Moderator Recovery System
- o Air supply and filter breakthrough alarms for Airborne activity Confinement System
- o Emergency Cooling System magnetic flow meters
- o Control Room habitability
- o Normal power

Verification of the operability of systems (completion of the Technical Specification Surveillance Requirements and verification of the procedures and operability of the above systems that are not yet part of the Technical Specifications) shall be documented in the WSRC Request for Startup Authorization.

11.2.4 Conclusion

The Technical Specifications (WSRC-TS-10003 Revision 0) are acceptable for chargeback and shutdown conditions. WSRC will complete and verify the Phase 1 and Phase 2 Technical Specifications and submit a revision prior to restart. This is an open item. DOE review and approval of this revision is required prior to startup.

The Phase 1 and Phase 2 Technical Specifications were submitted to DOE for approval as WSRC-TS-10003 Revision 0 on December 18, 1990 (Reference 5). At DOE request, WSRC included a new Technical Specification for the Ultimate Heat Sink. In the submittal, WSRC noted that the review verified the Limiting Conditions for Operation, and the associated Surveillance Requirements applicable to chargeback and shutdown. DOE approved Revision 0 on January 25, 1991, (Reference 6). Until WSRC-TS-10003, Rev. 0, is fully implemented, WSRC will comply with the existing DPST-TS-105; after implementation of WSRC-TS-10003, DPST-TS-105 will be withdrawn for the K-Reactor. Additionally, there are a number of open items remaining that relate to startup and operations, including:

- o blanket gas pressure
- o reactivity requirements - (after shutdown)
- o margin of control
- o scram circuit setpoints, allowable values, and response times
- o safety computer
- o process water system Bingham Pump DC motors
- o ECS parameters

These open items and the related safety analyses are expected to be resolved by May 1, 1991, and will be included in the next planned revision to the Technical Specifications. An emergent Technical Specification change request was submitted to DOE on March 11, 1991 (Reference 7). This change allows both GM diesels to be inoperable when there are no heat generating assemblies in the reactor. The request was approved by DOE on March 15, 1991 (Reference 8) as Revision 1.

Revision 0 of WSRC-TS-10003 contains bases for the new and reformatted Technical Specifications. The bases are at least as complete as those in DPST-TS-105 and the Technical Standards, and a reference to the appropriate Safety Analysis Report section has been added as requested by DOE. WSRC will verify the bases as part of Phase 3 of the TSIP. DOE is involved in the review and approval of the SAR, and the bases represent a consolidation of previously approved information and information that will be approved prior to restart. The WSRC-TS-10003 Revision 0 bases are acceptable for chargeback and shutdown conditions.

Phase three of the Technical Specifications Improvement Program will use the NRC's Interim Policy Statement on Technical Specification Improvement. These criteria are being used by the commercial industry to evaluate technical specifications and determine whether a specification should be removed and relocated to other documents. WSRC's use of these criteria is based on the Westinghouse Owners Group's Methodically Engineered, Restructured and Improved Technical Specifications (MERITS) Program, which is under review by the NRC.

11.2.3 Evaluation

The TSIP calls for a three-phase plan to provide reactor-specific Technical Specifications equivalent to those required by NRC for commercial power reactors. This plan is as follows:

- o The first phase of the TSIP includes 13 new Technical Specifications. These include Limiting Conditions for Operations (LCOs) and Surveillance Requirements (SRs).
- o The second phase of TSIP requires the development of Technical Specifications that correspond to items already addressed by the existing documents.
- o The third phase of the TSIP requires a revision to the Technical Specifications and Bases, in accordance with the NRC's Interim Policy Statement on Technical Specification Improvement (52FR3788, February 6, 1987).

WSRC has provided the TSIP (Reference 2) and a "Technical Specification Interim Program Plan" (Reference 3). The interim plan was approved by the DOE staff on February 5, 1990.

The restart criteria for the administrative controls portion of the Technical Specification were also based upon practices in the commercial nuclear industry. The Nuclear Safety Control Procedures (NSCP) in commercial reactors were replaced by Administrative Control Technical Specifications (ACTS). The ACTS for SRS reactors were first issued in 1989. Revision 1 to the ACTS was issued on January 12, 1990, and was approved by the staff with comments on March 12, 1990. Staff comments and additional WSRC self-review comments generated Revisions 2 and 3 to the ACTS. Revision 2 dealt with Temporary Procedure Changes which have been approved. Revision 3 to the ACTS incorporated many general changes throughout the document, and was approved by DOE on November 16, 1990 (Reference 4).

The ACTS are comparable in scope to those in commercial reactors and include additional requirements applicable to the DOE SRS reactors. The ACTS have been incorporated into Revision 0 of WSRC-TS-10003 (Reference 5).

As an interim measure for other Technical Specifications, a Linking Document was issued to assist in linking Technical Specifications, Technical Standards, and Operating Procedures to ensure that operators would have available a method to demonstrate, through observable plant parameters, that the facility was being operated within the bounds of the safe operating envelope. The present Technical Specifications (Revision 0) incorporate the relevant information from the Linking Document.

Where required, the bases for Limiting Safety System Settings will be included in a setpoint document (ROMP Item IM-74). DOE approval of this document is required prior to the approval of the final Technical Specifications revision prior to restart.

The bases for the set points shall be provided.

(c) Limiting Conditions for Operation (LCO) and Surveillance Requirements (SR)

Limiting Conditions for Operation shall provide the lowest functional capability or performance levels of equipment required for safe operation of the facility. Where applicable, LCOs shall specify limits for parameters or conditions of equipment required for safe operation.

The Technical Specifications shall indicate that, when a Limiting Condition for operation is not met, the contractor shall take the appropriate action permitted by the Technical Specifications, which may include shutting down the reactor.

Surveillance requirements relating to test, calibration, and inspection shall provide acceptable assurance that the necessary quality of systems and components is maintained, that facility operation will be within SLs, and that the LCOs will be met. Surveillance requirements shall include applicable surveillance performance standards for equipment required for safe operation.

To the extent applicable to the SRS reactors, LCOs, and SRs shall address: reactivity control, power distribution, instrumentation, reactor coolant systems, emergency core cooling system, confinement system, plant systems, electrical power systems, charge and discharge, and special test exceptions.

The bases for the LCOs and SRs shall be provided.

(d) Design Features

Design features of the facility (such as materials of construction and geometric arrangements) which, if altered or modified, could have significant effects on safety and which are not covered by the SLs, LCOs, or SRs shall be incorporated in the design features section of the Technical Specifications.

(e) Administrative Controls

Administrative controls shall ensure adequate provisions relating to organization, staffing requirements, procedures, training, record keeping, review and audit, responsibility, radiation protection program, respiratory protection program, high radiation protection area, and violation reporting necessary in support of the Technical Specification implementation.

- (b) Phase 2 shall consist of the development of Technical Specifications covering subject material that corresponds to current Technical Standards. These Technical Specifications shall be comparable in content and format to those found in the commercial industry, except the bases will generally be included by reference to existing documents.
 - (c) Phase 3 is a post-restart criterion. This phase shall consist of the completion of the Technical Specification Improvement Program, which shall ensure that all Technical Specification upgrades are finished by January 1992 and that priority is given to resolving the more serious technical deficiencies first.
- (2) All elements in these three phases shall satisfy the following criteria for restart.

(a) Definitions

To the extent applicable without revising old documents, WSRC shall provide a list of definitions that includes the following: Technical Specifications, Safety Analysis Report, Technical, Transient Protection, and Confinement Protection Limits (DPSTM.110). Procedures (DPSOPs and DPSOLs) and other controlled documents shall ensure consistent interpretation in usage, with plant-specific terms clearly defined.

(b) Safety Limits (SL), Limiting Safety System Settings (LSSS)

Safety limits, which are limits placed upon important process variables that are necessary to reasonably protect the integrity of certain physical barriers guarding against the uncontrolled release of radioactivity, shall be provided.

The Technical Specifications shall require the reactor to be shut down if any safety limit is exceeded, and operation shall not be resumed until authorized by DOE.

The bases for the safety limits shall be provided.

Limiting safety system settings shall be specified for automatic protective devices for those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

The Technical Specifications shall indicate that, if the automatic safety system does not function as required during operation, the contractor shall take appropriate action (which may include shutting down the reactor) and DOE shall be notified of the cause of the condition and corrective action taken to preclude recurrence.

11.2 Technical Specifications

11.2.1 Background

The Savannah River Site (SRS) reactor operation has traditionally been controlled by the instructions in detailed operating procedures. In 1975, safety requirements were established in the Technical Specifications and more specific controls were established in the Technical Standards to identify overall plant safety requirements, which were previously determined by supervisors. The supervisor was responsible for understanding and interpreting the technical bases supporting these documents, which were described in other safety documents such as technical reports and technical manuals. It was determined that this process was deficient in human factors considerations, that it lacked an integrated approach, and that the validity of some of the technical bases were not well established and documented.

In early 1988, a Technical Specification improvement program (TSIP) (Reference 1) was initiated. The program was modelled after the Nuclear Regulatory Commission's (NRC) TSIP program. The goal of this program was to develop specifications for systems and equipment based upon observable parameters that preserve the operating envelope defined in the safety analyses.

11.2.2 Restart Criteria

The criteria for determining the adequacy of the SRS technical specifications for restart are described below.

- (1) WSRC shall establish a three-phase TSIP that provides reactor-specific Technical Specifications equivalent to those required by the NRC for commercial power reactors. The program shall meet the following criteria:

- (a) Phase 1 shall consist of the development of "new" Technical Specifications, including limiting conditions for operation and surveillance requirements, as listed below.

- Safety Rod Insertion Limit
- Safety Rod Operability
- Reactivity Requirements - shutdown
- Axial Power Distribution
- Radial Power Distribution
- Full Length Control Rod Positioning
- Partial Length Control Rod Positioning
- Supplementary Safety System - Operating
- Process Water System Leakage Detection Instrumentation
- Process Water System Operational Leakage
- Reactivity Anomalies
- Temperature Coefficients of Reactivity
(Coolant, Moderator, and Overall)

mixtures of tritium should be addressed. This is an open item.

11.1.4 Conclusion

WSRC has committed to completing analyses and providing an upgrade of Chapter 15 of the SAR. Formal, complete Chapter 15 analyses have not yet been provided by WSRC. Upon resolution of the open items identified above, the staff will be satisfied that the restart criteria have been met. Resolution of the open items will be documented in a supplement to the SER.

11.1.5 References

1. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Report of Nuclear Power Plants," Rev. 3, 1978.
2. DuPont, DPSTSA-100-1, "Safety Analysis of Savannah River Production Reactor Operation," October 1987.
3. WSRC, WSRC-SAR-RX-K, -L, -P, (WSRC-RP-89-741, -742, -743), "Savannah River Site Production Reactor Safety Analysis Report," 1989.
4. WSRC, WSRC-RP-89-797, "Action Plan for Updated Chapter 15 Accident Analysis in the SRS Production Reactor SAR" (U), November 15, 1989.
5. WSRC, WSRC-RP-90-10, "Scope of Design Basis Events, Method of Assessment, and Acceptance Criteria for Updated Chapter 15 Accident Analysis in the SRS Production Reactor SAR" (U), January 1990.
6. WSRC, WSRC-RP-89-373, "Action Plan to Upgrade Thermal-Hydraulic Limits for SRS Reactors" (U), June 30, 1989.
7. WSRC, WSRC-RP-89-1296, "Supplement to the Action Plan to Upgrade Thermal-Hydraulic Limits for SRS Reactors" (U), November 30, 1989.

operator action required during the initial 10 minutes of the event. Simulator training has been planned and is outlined in the action plan. The Updated SAR Chapter 15 Action Plan states that operator actions will be considered. WSRC could not identify any non-LOCA DBEs that would require operator action under stress. Additional effort is required to identify transients and accidents requiring operator action as well as in modeling time delay of operator action in the DBE analyses. Compliance with the restart criterion concerning operator action is an open item.

The Action Plans do not explicitly address the requirement that the core can be maintained at an acceptable temperature for an extended period of time after ECS initiation. WSRC indicated that preliminary experimental work on maintaining long-term core cooling has been completed. More experimental work is planned to see if the flow can be safely reduced. This is an open item.

The effects of a postulated accident on other systems should be considered in the analyses. None of the Action Plans addressed this specific requirement. However, flooding of AC/DC pumps and motors is included in the consideration of the cooling water LOCA effects on the process water LOPA. SER Section 6.3.2 addresses environmental effects of postulated accidents. SER Section 6.3.1 addresses the loss of normal electric power event scenarios. WSRC should analyze coincident Loss of Normal Power in conjunction with some DBEs in Chapter 15. The effects of postulated accidents on other systems is an open item.

The fuel clad integrity requirement would be met if WSRC can demonstrate that no fuel damage precursor conditions occur. Both Action Plans indicate that thermal hydraulic limits (FI and ECS), which preclude reaching fuel damage precursors, will be applied to all DBEs. Maintenance of fuel clad integrity is an open item.

Neither of the Action Plans explicitly addresses the requirement of considering radioactive releases due to failure of liquid-containing tanks and overflow to the 50 million gallon retention basin. This is an open item until the completed analyses address this restart criterion in an acceptable manner.

Regarding fuel design limits, both Action Plans addressed the uncertainties in the analyses. This is an open item until the analyses are completed.

The assessments of the reactor vessel roll anchor loads and confinement responses to ECS actuation, both due to pressurization, were not specifically covered in either Action Plan. WSRC was requested to provide these assessments. PWS pressure assessment is an open item.

The core should remain subcritical and/or amenable to cooling considering calculated changes in core geometry. The staff considers that if the fuel-clad integrity requirement is met, this requirement would also be met. The loss of fuel accident postulated by WSRC will require special consideration.

The hydrogen concentration requirement was not addressed explicitly in the Action Plans. The staff considers that hydrogen release would be inhibited if WSRC can show that no DBEs will result in fuel damage. Potential hazardous

WSRC is committed to providing sufficiently detailed information in the SAR to form the basis for technical specifications. This remains an open item.

WSRC has committed to perform radiological release calculations. These are expected to confirm that the site boundary releases conform with Title 10 of the Code of Federal Regulations (CFR) Part 100. Conformance with 10 CFR 100 requirements is an open item.

DOE is committed to bring the SAR up to commercial standards. Preparation of an Action Plan for post-restart Chapter 15 upgrades was committed to by the non-LOCA WSRC/SAG section; whereas, no such commitment has been made from the Power Limits Program under WSRC/NES. A written commitment from the LOCA section, and a schedule from both sections that shows the action plans will be prepared by January 1, 1992, are needed to close this open item.

Specific Restart Criteria

The restart criteria dictate that the most limiting plant systems' single failures be identified and assumed in the analyses of Chapter 15. DOE has provided guidance to WSRC concerning the methodology to use to identify single failures. This methodology would require that the worst credible single failures be properly included in the analyses. Conformance with this criterion is an open item.

The restart criteria require that transient and accident analyses include the assumption that mitigating systems are not actuated until the most conservative setpoint is reached. The Action Plans discuss setpoints and the time delays allowed for scram. In responding to the staff's inquiry, WSRC has indicated that worst-case setpoints are assumed for the analyses and, in particular, for the scram or the actuation of the Supplementary Shutdown System (SSS). WSRC further assured the staff that it has prepared specific calculations containing tables for all trip setpoints used for its analyses and that WSRC is currently developing a setpoint methodology document. Determination of setpoints is an open item.

In regard to the requirement of conservative scram characteristics, WSRC has indicated that, due to previous experience and operational procedures, 9 out of 66 safety rods will be assumed ineffective during scram. This assumption would satisfy the restart criterion. Until the analyses are completed, this is an open item.

The analyses for the SAR Chapter 15 shall be based on the most limiting combination of neutronic and thermal hydraulic parameters. The Action Plans did indicate that conservative assumptions will be used. The staff has concluded that if neutronic and thermal hydraulic parameters used in the analyses are based on the most restrictive, but consistent, combination of temperature coefficients and neutron flux shapes over the expected core life, the requirement would be met. This is an open item.

A critical requirement for the acceptance of transient and accident analyses is the consideration of credibility of operator action under stress. The Thermal Hydraulic Limits Action Plan indicated that only during a LOPA is

from the moderator during a LOCA is expected. ATWS considerations for non-LOCAs will be included in the applicable DBE sections.

The final draft SAR Chapter 15 indicated that WSRC has included events involving radioactive release and has addressed ATWS in the Rotovalve closure and the small break LOCA DBEs. The adequacy of radioactive release and ATWS considerations are open items.

The restart criteria require that DBEs be categorized according to expected occurrence frequency. Higher frequency events should have larger margins of safety. From the Action Plans, the staff determined that this requirement will be satisfied. Proper treatment of the DBEs according to expected frequency is an open item.

All DBEs are to be analyzed for their responses to postulated disturbances and failures of one system or component important to safety. DOE has provided guidance in the identification and assessment of single failures, and WSRC is implementing a systematic procedure to identify single failures. The disposition of single failures that have been identified by WSRC remains an open item.

The staff has reviewed a standard format outline for SAR Chapter 15, which was prepared by WSRC. This format conforms to RG 1.70 and will be applied to all transient and accident analyses.

The restart criteria require that the analyses shall meet the acceptance criteria in applicable sections of Chapter 15 of NUREG-0800, as adapted to the SRS reactors. From the Action Plans, the staff determined that this criteria is satisfied.

The certification of all analytical methodologies for Chapter 15 analyses, which require the use of computer programs, will be performed in accordance with Section 5.8 of this SER. These analytical methodologies have been, and will be in the future, the subject of many WSRC internal peer reviews and DOE independent reviews. Completion of a DBE analysis requires the preparation of supporting calculations. WSRC calculations are reviewed by at least one analyst and are approved by the responsible manager according to WSRC's internal QA procedure. The staff determined that the approach being utilized by WSRC is acceptable. However, pending completion of the analyses, this remains an open item.

The specific operational limits originally developed for the Upgraded SAR Chapter 15 were the flow instabilities (FI) and emergency coolant system (ECS) limits. These limits are based on extensive full-scale experiments conducted at various national organizations, on supplementary analytical studies, and on the application of numerous computer codes which have been developed by organizations inside and outside of SRS. These tests and computer codes are described in the action plans. The staff determined that the programmatic requirement for the establishment of specific operation limits and correlations has been met; however, the determination of the complete set of operational limits is an open item due to several incomplete accident analyses.

11.1.3 Evaluation

Action Plan Review

WSRC has provided DOE with a detailed Action Plan (Reference 4) for updating Chapter 15 of the SAR. This Action Plan was prepared for all of Chapter 15. The DBEs were divided into two groups: the first group emphasized the transients due to changes in core reactivity (non-LOCA events); the second emphasized the transients in coolant flow (mainly LOCA events). Although the list of DBEs is complete, only non-LOCA DBEs were specifically addressed in subsequent implementation guidance (Reference 5). The staff has completed review of this guidance and has determined that both the methodology and objectives are reasonable. It should be noted, however, that the input data, methodology, and acceptable design limits for the DBEs that result in a LOCA are not addressed in this guidance, but rather under other action plans prepared for upgrading the thermal-hydraulic limits (References 6 and 7). Also, the problems of gamma heating and the potential of safety rod melting are not addressed in the thermal-hydraulic action plans. The LOCA analyses fall under the Power Limits Program, which is addressed in Section 11.3 of this report.

General Criteria

A broad spectrum of non-LOCA transients and accidents are to be analyzed by the contractor for the Updated SAR Chapter 15 analysis. In addition, specific analyses are to be provided for the Startup Accident and the Drop of a Partial Length Rod from the highest worth position. The staff examined the Chapter 15 Action Plan and concluded that these accidents are being addressed. The double-ended guillotine break (DEGB) LOCA, the loss of pumping accident (LOPA), a spectrum of breaks including the large break (LB) LOCA, the gang rod withdrawal (GRW), the pump shaft break (PSB), DBA accidents that are considered under the Power Limits Program, the startup accident, and the partial length rod drop accident were identified in Table I of the Updated SAR Action Plan, and in Table 1 and Chapter 4 of the Upgraded Power Limits Action Plan. Completion of an acceptable upgraded SAR Chapter 15 is an open item.

DOE requires that the assumptions, input data, and analyses following the Upgraded Thermal-Hydraulic Limits Action Plan and the Updated SAR Chapter 15 Action Plan should be consistent. Important parameters for thermal-hydraulic limits analysis are mostly flow related; whereas, for non-LOCA analysis, the important parameters are temperature related. Inputs to transient analyses may, therefore, require different sets of boundary conditions. The consistency requirement will be satisfied if any differences are specifically noted and can be satisfactorily explained in the SAR. Compliance with this requirement is an open item.

Restart Criteria for All Analyses

The DBEs are to be categorized according to one of eight types of resultant events, which are listed in the Restart Criteria section. Except for the radioactive release from a subsystem or component, none of the DBEs is expected to involve direct radioactive release, although some tritium release

- (e) If operator action is required during the initial 10 minutes of the transient, the operator actions shall be reviewed to ensure that they are credible, that they can occur within the necessary time frame, and that an acceptable confidence level shall be provided.
- (f) WSRC shall confirm that after any calculated successful initial operation of the emergency cooling system (ECS), the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.
- (g) Effects of the postulated accident on other systems shall be considered.
- (h) When loss of power is either an initiated event or a consequential event, an analysis of loss of offsite power (offsite power is defined in Section 6.3, "Electrical Systems") and its effect on the consequences of accidents shall be provided. When loss of power is either an initiated event or a consequential event, the analysis shall consider the effect of loss of offsite power on trips, initiation of auxiliary systems, and the sequence of events for the accidents.
- (i) WSRC shall confirm that fuel clad integrity will be maintained throughout anticipated transients.
- (j) The potential for radioactive releases due to failure of liquid-containing tanks shall be assessed. Radioactive releases due to accident mitigation, such as overflow to the 50-million-gallon retention basin, shall be considered.
- (k) The fuel design limits shall include allowances for uncertainties and modelling omissions.
- (l) The process water system pressure assessment shall include allowances for uncertainties and modelling omissions (for example, by evaluating roll anchor response to ECS venting and on evaluation of confinement protection limits as part of the power limits).
- (m) WSRC shall confirm that the calculated changes in core geometry are such that the core remains subcritical and/or amenable to cooling.
- (n) The calculated hydrogen (and isotopes) concentration during an accident shall not present an unacceptable hazard (for example, flammable or explosive mixture).

Content of Safety Analysis Reports for Nuclear Power Plants." The content of the accident and transient analysis shall meet the general guidance of RG 1.70, as adapted for the SRS reactor design.

- (e) The accident and transient analyses shall meet the acceptance criteria in applicable sections of Chapter 15 of NUREG-0800, "NRC Standard Review Plan," as adapted to the SRS reactors.
- (f) Analytical methodology for transient analysis of DBEs shall be certified according to the criteria in Section 5.8 of this SER. Modeling of DBEs with the methodology shall include appropriate uncertainties, and the output shall be checked and independently verified per appropriate Quality Assurance (QA) procedures.
- (g) Specific operational limits and correlations shall be demonstrated by experiments, tests, or analyses to be conservatively safe and applicable to the system design and the transient or accident category under consideration.
- (h) The information provided in the SAR shall be sufficiently detailed to form the basis for technical specifications.
- (i) WSRC shall ensure that in the event of a design basis accident site boundary releases do not exceed those permitted by 10 CFR 100.
- (j) A program with an action plan to upgrade Chapter 15 of the SAR to meet standards comparable to commercial reactor standards shall be in place by January 1, 1992.

(3) Specific Restart Criteria

- (a) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR 50, shall be identified and assumed in the transient and accident analyses and shall satisfy the positions of RG 1.53.
- (b) Transient and accident analyses shall assume that mitigating systems are not actuated until the most conservative (worst-case) setpoint is reached with allowance for instrument inaccuracy in accordance with RG 1.105.
- (c) Conservative scram characteristics shall be assumed (i.e., maximum time delay with the most reactive combination of operationally unavailable rods and possible individual rod insertion delays).
- (d) Core burn-up shall be selected based on the most limiting actual combination of fuel, target, coolant, and moderator temperature coefficients; void coefficient; Doppler coefficient; axial power profile; and radial power distribution.

- o a spectrum of breaks, including large break (LB) LOCA
- o gang rod withdrawal (GRW)
- o pump shaft break (PSB)

These analyses, which may not be as comprehensive as the limiting event analyses, shall include estimates of conservatism and shall be fully documented.

- (b) A detailed assessment of the startup accident and the partial length rod drop accident shall be documented.
- (c) The remaining Chapter 15 accidents may be analyzed with more qualitative logical arguments including consideration of the low restart power level and the bounding nature of the above transients. However, as a minimum, justification showing that each design base event (DBE) of Chapter 15 meets its applicable limits shall be documented.
- (d) Assumptions, input data, and analyses based on the Upgraded Thermal-Hydraulic Limits Action Plan and the Upgraded Chapter 15 Action Plan shall be consistent.

(2) Restart Criteria for All Analyses

- (a) A broad spectrum of initiating events shall be considered. Malfunctions or equipment failures shall be postulated, and postulated initiating event shall be assigned to one of the following types of events:
 - o increase in heat removal by the cooling water system
 - o decrease in heat removal by the cooling water system
 - o decrease in reactor coolant system flow rate
 - o reactivity and power distribution anomalies
 - o increase in reactor coolant inventory
 - o decrease in reactor coolant inventory
 - o radioactive release from a subsystem or component
 - o anticipated transients without scram
- (b) Each initiating event shall be categorized according to its expected frequency of occurrence.
- (c) Analyses of the response of the plant to postulated disturbances in process variables and to postulated malfunctions or failures of equipment shall be provided to demonstrate that the plant satisfies requirements established for the health and safety of the public and employees and for protection of the environment.
- (d) Consistent with the general acceptance criteria, documentation of the accident and transient analyses shall be equivalent in detail to that specified in RG 1.70, Chapter 15, "Standard Format and

11 OPERATING LIMITS

11.1 Transient and Accident Analysis

11.1.1 Background

Chapter 15 of Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," LWR [Light Water Reactor] Edition (Reference 1) contains specific requirements regarding analyses to be performed for specific operational occurrences and postulated accidents. When Dupont submitted its safety analysis report (SAR) (Reference 2) for the Savannah River Site in October 1987, it was determined that the information provided in Chapter 15, "Accident Analysis," was not consistent with the general format and content required by Regulatory Guide 1.70. In addition, the information and documentation provided were not sufficient to support specific analyses of operational occurrences and postulated accidents. In 1989, Westinghouse Savannah River Company (WSRC) submitted a revised safety analysis (Reference 3). Both the format and content of Chapter 15 were much improved in the WSRC submittal; however, the analyses were still deficient.

The DOE staff found that WSRC had provided "typical" and/or "sample" data, rather than the specific and/or appropriate bounding data required by RG 1.70. The staff also found that WSRC's descriptions of the Chapter 15 transients did not include pertinent required data, such as initial conditions, key nuclear and mechanical parameters, safety system setpoints, instrument errors, and response times.

Because of concerns regarding the deficiencies of the Chapter 15 analyses, WSRC must upgrade these safety analyses to meet commercial nuclear industry standards for similar analyses. Prior to reactor restart, the DOE staff will review the upgraded transient and accident analyses for completeness of analysis of events and consequences, including the conservative consideration of human performance and equipment designed to mitigate the radiological consequences of credible accidents.

11.1.2 Restart Criteria

The criteria for determining the adequacy of transient and accident analyses for restart are described below.

(1) General Criteria

The restart criteria listed below will be applied to detailed, explicit accident and transient analyses for the most limiting events. Two of these events are: (1) the double-ended guillotine break (DEGB) loss-of-coolant accident (LOCA), which is the most restrictive LOCA, and (2) the loss-of-pumping accident (LOPA). Detailed analyses of these events shall be documented in a pre-restart upgrade to the SAR.

- (a) The bounding nature of these events shall be confirmed by supporting analyses of:

10.16.4 Conclusion

The staff has identified open items in the operating experience feedback program as discussed above. Upon satisfactory resolution of these open items, the staff will be able to conclude that the operating experience feedback program is acceptable for restart. Resolution of these open items will be reported in a supplement to the SER.

10.16.5 References

1. Institute of Nuclear Power Operations, INPO 85-001, "Performance Objectives and Criteria for Operating and Near-term Operating License Plants," Rev. 1, April 1987.
2. _____, INPO 85-019, "Industry Operating Experience Review," Rev. 1, April 1987.
3. _____, INPO 85-020, "In-House Operating Experience Review," Rev. 1, February 1987.
4. WSRC, RDP 9.06, "Operating Experience Review (U)," (draft).
5. WSRC, RDP 9.02, "Occurrence Reporting System," Rev. 0, November 15, 1990.
6. WSRC, WSRC-8Q15 RSES 301, "Outside Experience Review," Rev. 1, August 6, 1990.
7. WSRC, RDP 9.05, "Corrective Action Process for Reportable Occurrences," Rev. 0, December 1, 1990.

RDP 9.06 requires that relevant internal and external operating experiences be reviewed as part of the evaluation of significant internal events. In addition, RDP 9.06 requires that significant internal events receive a second multidisciplinary review by the Unusual Occurrence Review Committee (UORC) to ensure that all concerns are addressed.

Internal events will be trended to identify recurring problems and to determine corrective actions. RDP 9.06 requires the Manager, Plant and Human Performance, to trend these items and report the results to management on a periodic basis.

Post-trip reviews will be evaluated to verify that they are comprehensive and include identification of the root cause(s) of the trip. Identification and resolution of discrepancies between actual and expected plant responses to the trip and documentation to support results and recommendations will also be evaluated. These evaluations will be documented in SER Section 9.3.

RDP 9.06, step 4.1.17 requires the Manager, Plant and Human Performance, to ensure that pertinent internal operating experience information is distributed to appropriate personnel and departments and that recommended corrective actions are carried out in a timely manner.

The procedures ensure external operating experience that has been determined to be applicable will receive a comprehensive evaluation and appropriate consideration will be given to the root causes and generic implications. Appropriate corrective action will be completed in a timely manner. RSES 301 states that if an operating experience is found to be applicable, a report is developed by the evaluator and transmitted to the appropriate personnel for action.

Attachment 1 to RSES 301 lists a number of documents that should be reviewed as sources of significant external operating experience information. Examples of the documents listed include DOE Unusual Occurrence Reports (UORs); INPO Significant Operating Experience Reports (SOERs); INPO Significant Event Reports (SERs); INPO Significant by Others (SOs); NRC letters, bulletins, and information notices; and vendor bulletins.

RSES 301, Attachment 2, RSES Outside Experience Screening Form, contains guidance to ensure that checks will be performed to verify that external operating experience information is being properly classified for applicability. The form provides for review by two individuals. The review process involves an initial review for applicability, management review, and line management review.

RDP 9.06 addresses many of the issues identified in the restart criteria. However, the procedure has not yet been approved. Pending approval of this procedure and follow-up review of the issues addressed by this procedure, the approval of this procedure will be tracked as an open item.

external events. These levels are "Immediate Notification," "Applicable Information," and "Information Only." The internal review program, however, contains no administrative controls to ensure that internal operating experiences are screened to ensure that plant personnel do not routinely receive extraneous and unimportant information.

The DOE staff has determined that the internal review program does not fully meet the restart criteria in that no administrative controls are established to ensure evaluated internal operating experiences are screened prior to distribution, ensuring plant personnel do not routinely receive extraneous and unimportant information. Pending revision to the internal operating experience review procedures to provide a screening, this issue remains an open item.

Administrative controls should also be established to provide suitable checks to ensure that conflicting or contradictory information is not conveyed to operators and other personnel until a proper review is completed. Review of this information is completed through Attachment 2 instructions in RSES 301.

The DOE staff has determined that the operating experience review program does not fully meet the restart criteria in that no administrative controls are established for the internal review program to ensure that operating experience reviews do not convey conflicting or contradictory information to operators prior to a proper review. Pending revision of the internal operating experience review procedures to provide these controls, this issue remains an open item.

The Operating Experience Review Organization Plan ensures that an effective evaluation will be performed periodically to appraise management of how well the operating experience program is functioning at all levels and if it is attaining desired results. RDP 9.06 assigns the Vice President and General Manager, RRD, the responsibility of ensuring effective operation of the division's OER program.

The Plan requires management to be appropriately involved in operating experience review activities to assure adherence to policies and procedures and to identify and correct problems. RSES 301 requires that all transmittal letters be signed by the RSES Manager or designee. RDP 9.06 lists specific responsibilities for all persons involved in the operating experience review up to and including the Vice President and General Manager, RRD. The procedure specifically addresses effectiveness reviews for the overall program in accordance with the Reactor Operations Assessments Performance Assessment Program.

The requirement for the program to ensure internal events will be screened for significance and prioritized prior to evaluation is addressed in RDP 9.06, which requires the Shift Manager to perform this screening.

The procedures also ensure rigorous evaluation will be performed on significant internal events to determine root causes, generic implications, and necessary corrective actions to prevent recurrence.

The Operating Experience Review Organization Plan should ensure that action items resulting from operating experience reviews will receive appropriate approval and will be tracked to completion. RSES 301 states that transmittals requiring a response will be tracked by the RSES Commitment Tracking System per Procedure 102 and that follow-up transmittals will be issued every 3 months until a response is received from the responsible organization. In addition, the procedure states that the items will be tracked to completion. RDP 9.02 states "Track the implementation of corrective actions committed to the Final UO Reports."

The Operating Experience Review Organization Plan ensures that the review results are incorporated into the training program, procedures, or operating instructions. RSES 301 does not specifically require that review recommendations address the training program, procedures, or operating instructions. The procedure does, however, state that the review recommendations are transmitted to the affected organizations and tracked until a response is obtained. In addition, the manager issuing the transmittal letter requiring closure is responsible for concurrence with the adequacy of the closure. RDP 9.05, "Corrective Action Process for Reportable Occurrences" (Reference 7), also has no specific requirements for changing the training program, procedures, or operating instructions; however, extensive tracking and implementation activities are included in the procedure. These activities include statusing, tracking, assignment of responsibilities, and methods of implementation escalation if the implementation schedule is not met.

In order to ensure the operating experience review information is properly and consistently disseminated to the appropriate organizations and personnel, program procedures should identify recipients of various categories of information. RSES 301, Attachment 1, lists the responsible review groups for specific types of outside information. The minimum distribution of evaluation reports and/or recommendations is found on Table 1 of that procedure. RDP 9.06 requires the Manager-Plant and Human Performance to distribute the reports to the affected organizations, the DOE facility representative, and the Manager-Reactor Safety Assessment.

Significant operating experiences should be disseminated to the affected personnel in an expedited fashion so they can become aware of and understand information that should not wait for emphasis through routine training/retraining or other routine plant actions. Assurance that significant operating experiences will be properly disseminated is found in RSES 301. The procedure specifically addresses immediate notification of affected personnel through initial oral reports followed by an official transmittal. Additionally, RDP 9.06 requires the Operations Manager to deliver a completed notification package to the Manager-Plant and Human Performance within 24 hours, who in turn distributes reports to the DOE facility representative and the Manager-Reactor Safety Assessment.

Administrative controls are necessary to ensure extraneous and unimportant information on operating experiences is not received such that priority information is obscured, detracting from overall program job performance and proficiency. RSES 301 provides three levels of information transmittal for

- (c) Checks are performed to verify that external operating experience information is being properly classified for applicability.

10.16.3 Evaluation

Operating experience is reviewed in accordance with RDP 9.06, "Operating Experience Review (U)" (Reference 4). RDP 9.06 is the administrative procedure that describes the overall Operating Experience Feedback Program and the inter-relationships between the various program areas. Internal operating experience is reviewed in accordance with RDP 9.02, "Occurrence Reporting System" (Reference 5). External experience reviews are accomplished through RSES 301, "Outside Experience Review" (Reference 6).

The required staffing and resources for the operating experience review program are provided by the implementing procedures. For outside experience reviews, RSES 301 defines the responsibilities for the Reactor Safety Evaluation Section (RSES) Manager, Reactor Safety Assessment (RSA) Manager, Reactor Regulatory Compliance (RRC) Manager, Outside Experience Review Coordinator, Document Reviewer/INPO Nuclear Network Reviewer, and Reactor Safety Assessment Secretary. RDP 9.06 lists the responsibilities for line management, Shift and Operations Managers, Manager - Plant and Human Performance (P&HP), Manager - RSE, and the Deputy General Manager, RRD. RDP 9.02 lists the responsibilities for individuals and management associated with the development of occurrence reports.

The operating experience review program should provide sufficient guidance to ensure that interfaces with internal and external support groups are clearly defined and understood. RSES 301 describes the receiving of information and distribution of reports to affected WSRC departments and to the INPO Nuclear Network. In addition, Table 1 provides a specific list of internal and external organizations that will be issued copies of transmittals. RDP 9.06 describes within the procedure the interfaces between the individual managers, DOE, Manager-Reactor Safety Assessment, Manager-Plant and Human Performance, Operations Manager, and the Manager-Standards and Assessment. RDP 9.02 provides a flow chart in Attachment A that graphically depicts the interfaces between various groups and individuals.

Specifying the technical experience and training requirements for personnel associated with the operating experience review program will ensure that personnel involved possess the required knowledge and skills to perform assigned tasks. The technical experience/training requirements for RSES 301 are deferred to RSES 109, Attachment 7. RSES 301 does state that OER document reviewers and INPO Nuclear Network reviewers shall also be trained in the use of RSES 301. RDP 9.06 requires the Manager-Plant and Human Performance to ensure that reviewing personnel have the technical experience and training to perform the assigned tasks.

RDP 9.06 requires the Manager-Reactor Safety Evaluation to ensure that review personnel have the technical experience and training to perform the assigned tasks. This procedure reiterates that requirement for Level II Managers. RDP 9.02 lists specific technical and training requirements for evaluation personnel.

- (a) Internal events are screened for significance and prioritized for evaluation.
 - (b) Rigorous evaluation is performed on significant internal events to determine root causes, generic implications, and necessary corrective actions to prevent recurrence.
 - (c) Relevant internal and external operating experiences are reviewed as part of the evaluation of significant internal events.
 - (d) Significant internal events receive a second, multidisciplinary review to ensure that all concerns have been addressed.
 - (e) Internal events are trended to identify recurring problems and determine appropriate corrective actions.
 - (f) Post-trip reviews are evaluated to verify they are comprehensive and include the following:
 - o identification and resolution of the root cause(s) of the trip
 - o identification and resolution of discrepancies between actual and expected plant responses to the trip
 - o documentation to support results and recommendations
 - (g) Pertinent internal operating experience information is distributed to appropriate personnel and departments and recommended corrective actions are completed in a timely manner.
- (3) WSRC shall establish and implement procedures for the review of external (outside SR reactors) operating experiences that meet the following criteria:
- (a) External operating experience that has been determined to be applicable shall receive a comprehensive evaluation and appropriate consideration shall be given to root causes and generic implications. Appropriate corrective action is completed in a timely manner.
 - (b) Sources of significant external operating experience information reviewed for applicability include the following:
 - o DOE Unusual Occurrence Reports (UORs)
 - o INPO Significant Operating Experience Reports (SOERs)
 - o INPO Significant Event Reports (SERs)
 - o INPO Significant By Others (SOs)
 - o NRC letters, bulletins, and information notices
 - o Vendor bulletins

- (f) Action items resulting from operating experience reviews receive appropriate approval and are tracked to completion.
 - (g) Procedures are to be established to incorporate the review results into the training program, procedures, or operating instructions.
 - (h) Recipients of various categories of information from operating experiences are to be identified by job title and/or job function to ensure proper dissemination of information.
 - (i) Significant operating experiences are to be disseminated to the affected personnel in an expedited fashion so personnel become aware of and understand information that should not wait for emphasis through routine training/retraining or other routine plant actions.
 - (j) Administrative control is to be established to ensure plant personnel do not routinely receive extraneous and unimportant information on operating experience such that it would obscure priority information or otherwise detract from overall job performance and proficiency.
 - (k) Administrative control is to be established to provide suitable checks to ensure that conflicting or contradictory information is not conveyed to operators and other personnel until the proper review is completed.
 - (l) An effectiveness evaluation is to be performed periodically to apprise management of how well the operating experience program is functioning at all levels and whether or not it is attaining desired results. This assessment examines, but is not limited to, the following areas:
 - o distribution of operating experience information to appropriate personnel
 - o technical reviews of significant operating experiences
 - o implementation of action items resulting from the review of operating experience
 - o use of data from internal events to improve plant safety and reliability
 - (m) Management is appropriately involved in operating experience review activities to ensure adherence to policies and procedures and to identify and correct problems.
- (2) WSRC shall establish and implement procedures for the review of internal (SR reactor) operating experiences which meet the following criteria:

10.16 Operating Experience Feedback

10.16.1 Background

Commercial nuclear utilities receive a steady flow of information on industry events from the Institute of Nuclear Power Operations (INPO), the Nuclear Regulatory Commission (NRC), vendors, and others. This information, along with information concerning in-house operating experiences, can help the utility prevent the recurrence of events that affect the safety and reliability of their nuclear generating stations. INPO 85-001, "Performance Objectives and Criteria for Operating and Near-term Operating License Plants" (Reference 1); INPO 85-019, "Industry Operating Experience Review" (Reference 2); and INPO 85-020, "In-house Operating Experience Review" (Reference 3), outline elements of a successful operating experience feedback program.

At the time of shutdown, the SRS operating experience feedback program was focused almost exclusively on internal operating experience. Operational events were documented; however, corrective actions were not incorporated in a timely manner. In addition, little attention was given to external operating experience.

WSRC committed to implement an operating experience review program in accordance with INPO 85-001, including: (1) the organization and administration to ensure program effectiveness; (2) the evaluation of in-house and industry operating experiences; and (3) the incorporation of lessons learned from operating experience into corrective actions to improve plant safety and reliability.

10.16.2 Restart Criteria

The criteria for determining the adequacy of the SRS operational experience feedback program for restart are described below.

- (1) The Operating Experience Review Organization and Administration shall meet the following criteria:
 - (a) The organizational structure is clearly defined.
 - (b) Staffing and resources are sufficient to accomplish assigned tasks.
 - (c) Responsibilities and authority of each management, supervisory, and professional position involved in the program are clearly defined and understood.
 - (d) Interfaces with internal and external supporting groups are clearly defined and understood.
 - (e) Operating experience review personnel have the technical experience and training to perform assigned tasks.

controls, circuit breakers) other than the controls in the control room that may indirectly affect the power level or reactivity of a reactor shall do so only with the knowledge and consent of the control room operators. The Conduct of Operations manual requires that manipulating or monitoring of the reactor controls must be performed by or under the direction of control room operators.

Requirements for training operations personnel on the access policy, professional behavior in the control room, and operation of plant equipment shall be addressed by the operator training program. The review of the operator training program is addressed in SER Section 10.2 "Training."

10.15.4 Conclusion

The Control Room Access/Decorum program review has been completed. The DOE staff has determined that the procedures now in place meet the restart criteria specified in the SER.

10.15.5 References

1. Institute of Nuclear Power Operations, INPO 85-017, Guidelines for the Conduct of Operations at Nuclear Power Stations, Chapter 3, "Control Room Activities," Rev. 01, April 1988.
2. WSRC, DPSOP-336, Reactor Operations Standing Instruction III.15, "CCR/KCR - Nonassigned Personnel Control," Rev. 1, April 1989.
3. WSRC, R 2.1, "Conduct of Operations Manual," Rev. 0, May 1990.

essential to supporting safe plant operation and activities authorized by shift management are to be conducted in the control room. The manual also states that potentially distracting activities such as non-work radio listening, game playing, and horseplay are prohibited in the plant. Non-job-related discussions are not to detract from operations personnel's professional responsibilities. Computers are required to be used only for company work activities. The Shift Manager is assigned the responsibility for maintaining control room decorum. The Shift Manager is also responsible for overseeing operating functions at the reactor to assure safe, efficient operation in compliance with DOE requirements, WSRC requirements, and WSRC commitments.

The Conduct of Operations Manual requires shift operations personnel to be attentive to system conditions and parameter trends by monitoring the information displays associated with each person's duties. It states that operations personnel should remain diligent and actively pursue resolution of safety concerns. Shift operations personnel are required to constantly remain alert and maintain awareness of plant status, anticipating conditions that could adversely affect plant reliability. The manual states that operators shall be alert and attentive to alarms and, if appropriate, shall take timely action to address and correct the cause of alarms. Additionally, it states that operators shall monitor the panel displays frequently to determine trends and to detect the development of problems. Operators are required to take prompt actions to determine the cause of the abnormalities, advise shift management of the abnormalities, and take corrective action.

The Conduct of Operations Manual requires that operators must not undertake more concurrent evolutions than they can safely and effectively control. It is stated that an excessive number of such concurrent evolutions can compromise the individual's ability to detect and respond to abnormal conditions.

The Conduct of Operations Manual addresses ancillary duties of the control room supervisor and control room operators. It states that it is the responsibility of the Shift Manager to limit the ancillary duties of the control room supervisor to prevent his/her distraction from primary responsibilities. Actions may include providing assistance to the control room supervisor or rescheduling certain activities. The shift manager is required to observe the control room supervisor periodically during periods of high administrative workloads or during periods of intense operations activity to determine that the ancillary demands on the control room supervisor's attention are not excessive. Similarly, the control room supervisor is required to monitor the volume of ancillary duties assigned to the control room operators and take action, if appropriate. Particular attention is to be paid to the number of personnel around the control panels.

The Conduct of Operations Manual, Section 3, "Operator Shift Responsibilities," defines the responsibilities of operators while on shift. It states that only a certified central control room operator or a supervisor holding certification may manipulate the controls of the reactor. Under certain conditions, trainees may also operate the controls. The manual states that any person who operates mechanisms and apparatus (e.g., valves, pump

(6) Operation of Control Room Equipment

Only those persons specifically authorized by reactor operations department administrative procedures shall operate control room equipment. When this equipment is operated by trainees, they shall be supervised and controlled by the operator who normally would perform the operations. WSRC shall have a formal policy to this effect and WSRC staff who would normally be expected to work in the control room shall be trained on this policy.

10.15.3 Evaluation

ROSI III.15 (Reference 2) provides the guidelines relative to entry into and work in the central control room (CCR) and crane control room (KCR) by nonassigned personnel. Nonassigned personnel are defined as persons other than assigned reactor shift crews.

Personnel are required by ROSI III.15 to use only the door to the shift clerk's office (fishbowl) for routine entrance and exit. This door allows entrance and exit from the common area between the CCR and the KCR. Other doors are for emergency exit only and are to be kept wire-sealed at all times. ROSI III.15 states that admittance to the CCR/KCR is for official business only, and only for business that must be conducted in the CCR/KCR.

Access to the control room area involves three different levels of access control (the shift clerk's office [fishbowl], the control room proper, and the "at the controls area"). The Conduct of Operations Manual (Reference 3) states that plant manager permission for access to the fishbowl is granted via authorization to the access computer code. With the proper code, the person may enter the fishbowl. Access to the control room proper is more restrictive. ROSI III.15 states that all nonassigned personnel must report to the Senior Supervisor prior to entering either the control room proper or the KCR. Access to the area near the control panels ("at the controls area") is more restricted; only those persons whose duties are necessary to support operations are permitted in these areas. Operations personnel whose work stations are the control room have access to these areas as necessary to perform their duties and carry out their responsibilities. The Conduct of Operations Manual provides access guidelines to these areas for operations and other plant personnel, including actions during emergency conditions or transients. The manual states that certain WSRC managers and the DOE Area Representatives are allowed to enter all areas of the control room without permission from control room personnel, although they are to check with CCR personnel to ensure their presence will not interfere with current or planned evolutions. All other personnel are required to obtain permission from control room personnel prior to entering.

The Conduct of Operations Manual, Section 2, "Conduct and Accountability," provides information on expected operator behavior, appearance, fitness for duty and alertness while operating the reactor. The manual contains requirements for shift operations personnel. These requirements include displaying professional and proper behavior at all times, reflecting a safety-minded approach to routine operations. Additionally, only activities

(3) Professional Behavior

- (a) Professional behavior shall be displayed in the control room at all times. Only activities essential to supporting plant operation and activities authorized by plant management shall be conducted in the control room. Potentially distracting activities such as radio listening, game playing, and horseplay shall be prohibited. Non-job-related discussions shall be minimized so as not to interfere with shift activities or the monitoring of plant parameters.
- (b) WSRC shall have a formal policy on control room decorum. WSRC staff who would be expected to regularly work in the control room, as well as their line management, shall be trained in that policy before restart. Line management responsible for maintaining the control room decorum shall be formally identified.

(4) Monitoring Control Panels

- (a) Operators shall be alert and attentive to control panel indications and alarms. Control panel indications shall be monitored frequently, and prompt actions taken to determine the cause and to correct abnormalities. WSRC management shall emphasize to operating staff the importance of closely monitoring and trending to detect problem situations early.
- (b) Operator response to alarms shall be timely, and actions shall be taken to address and correct, if appropriate, the causes of any alarms. All reasonable action shall be taken to clear alarming conditions. The number of evolutions affecting control panel indications that are performed concurrently shall be limited so that the operators' ability to detect and respond to abnormal conditions will not be compromised.

(5) Control Room Operator Auxiliary Duties

Duties assigned to control room operators shall not interfere with their ability to monitor plant parameters. Activities such as preparation of tagouts, reviews of operating procedures, required reading, and review of maintenance work activities shall not comprise a major portion of these operators' shift responsibilities, the administrative workload of the operators responsible for monitoring and operating the control panels shall be minimized. If one control room operator is involved in administrative tasks, other control room operators shall assume responsibility to monitor the unit. Some administrative activities are better performed away from the "at-the-controls" area by an operator who is not responsible for operating the control panel. Those activities shall be identified and assigned to operators or other persons who are not "at-the-controls."

10.15 Control Room Access/Decorum

10.15.1 Background

The control room is the most critical plant operating station and the coordination point for all important plant activities. Therefore, activities in the control room must be businesslike, and a professional atmosphere conducive to safe and efficient operation must be maintained. In addition, control room operators must not be overburdened with administrative responsibilities, and control room access must be limited so that operators will not be distracted from properly monitoring plant parameters.

At the time of shutdown, central control room requirements, such as professional behavior, monitoring the main control boards, control room operator ancillary duties, the operation of control room equipment, and access to the central control room, did not reflect commercial industry standards. The lack of professionalism and the constant distractions from telephones were contributing factors in the large pressure transient (water hammer) event that took place January 22, 1989.

10.15.2 Restart Criteria

The criteria for determining the adequacy of SRS control room activities for restart are described below.

(1) General

Control room activities shall be conducted in a manner that achieves safe and reliable plant operations in accordance with Chapter 3 of Institute of Nuclear Power Operations (INPO) document INPO 85-017 Revision 01, Guidelines for the Conduct of Operations at Nuclear Power Stations, "Control Room Activities" (Reference 1).

(2) Control Room Access

- (a) Control room access shall be limited to those persons who are on official business only. The "at-the-controls" area of the control room shall be clearly identified, and its boundary shall be understood by all persons who are granted access to the control room. Access to the "at-the-controls" area shall be restricted to persons who need to be in the area. Entry into this area shall be granted only to designated individuals. Persons who might need to enter this area should know who can grant access.
- (b) WSRC shall have a formal policy on control room access that clearly states who can grant access to the control room as well as the "at-the-controls" area of the control room. The policy shall also clearly state acceptable bases for granting access. Training on this policy shall be given to all persons with authority to grant access, all persons who regularly work in the control room and all persons who have been granted unescorted access to the reactor area.

3. WSRC, WSRC-RP-89-1146, STPM Section 5, "Modification Turnover and Transfer," Rev. 0, January 1990.
4. WSRC, WSRC-RP-89-1146, STPM Section 11, "Startup Test Program Manual Test Drawing Development," Rev. 0, April 1990.

RTD Standard Practice Manual, Section 8.1, "Design Authority Review" (Reference 2), states that CCR drawings that require revision will be entered into the Reactor Engineering Action Tracking System and that Design Change Forms will be initiated. This interim procedure establishes a review and tracking system for design and work packages affecting 46 systems, including all of the 13 listed in the restart criteria.

WSRC-RP-89-1146, Section STPM 05, "Modification Turnover and Transfer" (Reference 3), states that copies of the turnover packages are provided to the organizations responsible for CCR drawing updates. The procedure also states that the startup test engineer ensures that the CCR drawings have been updated and provides for the reactor project coordinator to request the system engineer to issue the updated CCR drawings.

Reactor operations personnel are being trained in the understanding and use of P&IDs through study guide CS-RO-IT-FUND-08-01-SG and cycle training manual RX-RST-084-SG-CT. The staff reviewed the training manuals and observed the training classes and found that both meet the training requirements of the SER.

In addition to the P&IDs being developed and issued to the control room, WSRC has developed a program for developing startup test diagrams. WSRC-RP-89-1146, Section STPM 11, "Startup Test Program Manual Test Drawing Development" (Reference 4), describes this program. This program calls for the development of as-built drawings for selected systems identified in the Startup Test Program for use by the test engineers in development of test procedures and performance of those test procedures. These as-built drawings were generated in one of two ways. Where controlled SRS drawings existed and were deemed adequate to support startup testing, these drawings were walked down to verify the as-built configuration. Where controlled SRS drawings were not adequate to support startup testing, test drawings were developed and walked down to provide the as-built configuration for use by the startup test personnel. These drawings are very similar in design and format to the P&IDs issued to the control room; however, they do not necessarily include all components, instrument loops, etc. that pertain to the system in question. Rather, they were developed based on the need to support startup testing and will encompass, as a minimum, all components, instruments, etc. necessary for the performance of a particular startup test.

10.14.4 Conclusion

The Control Room P&ID program review has been completed. The DOE staff has determined that the procedures now in place meet the restart criteria specified in the SER.

10.14.5 References

1. WSRC, DPSOP-336, Reactor Operations Standing Instruction No. 11.13, "Drawing Control," Rev. 0, February 1990.
2. WSRC, RTD Standard Practice Manual, Section 8.1, "Design Authority Review," Rev. A.

WISR-38 requires the establishment of a visual inspection and surveillance program to monitor the structural integrity of the CWS piping. This is addressed in Section 14.3.3 below.

14.3.3 In-Service Inspections (ISI)

Periodic Equipment Inspections

WSRC inspections for equipment and components are performed on the basis of a prioritized, selective program, the Periodic Equipment Inspection Program, DPSOL 105-1851B-PLK (Reference 9). This program was instituted under the previous contractor and is still in effect. Items are selected for inspection on an annual basis based on their importance to the plant's safe operation and shutdown capabilities. Inspection intervals range from semi-annual to 15 years. Components included in the current inspection programs are the PWS piping and weldments to the first isolation valve, the reactor tank, and related components. The Engineered Safety Features (ESF) and the Cooling Water System (CWS) are also included in this program.

During its review, the staff noted some inherent drawbacks in the inspection process. This "log-keeping" type inspection evolved at SRS as a response to problems experienced in day-to-day operation. The staff found there was an absence of specified acceptance criteria for the inspections carried out. Also, the procedure requires a written report to be submitted for each inspection and a copy to be filed in the Area History File. A copy is also to be sent to the Reactor Technology Department. However, this procedure did not state who is responsible for evaluation, acceptance, or origination of a nonconformance report. WSRC has implemented departmental procedures to resolve these deficiencies by including inspection acceptance criteria in the work packages. The staff determined that DPSOL 105-1851B, as supplemented by other inspection programs discussed below, forms an acceptable basis for restart.

Some of the items addressed in the in-service inspection program are also addressed in the "Augmented ISI Plan," DPSTM-88-100-1 (Reference 12), causing redundancy and confusion. WSRC is aware of this situation and is taking steps to correct the problem by gradually merging the two separate plans into one comprehensive program that doesn't have the shortcomings of the older, DPSOL-105-1851B plan. This corrective action is not a restart constraint and will be implemented as a part of the longer term ISI plan.

A detailed comparison of the existing ISI program to the requirements of the ASME Section XI has been performed by WSRC and included in the Restart Inspection and Testing Plan (Reference 8). The staff reviewed this comparison and found it satisfies the restart criterion.

Augmented ISI Plan for the Process Water System (PWS)

The development of a comprehensive ISI program for the PWS has been undertaken to enhance reactor safety by augmenting and extending the scope of the current PWS pressure boundary surveillance procedure, DPSOL 105-1851B. This program is described in DPSTM-88-100-1. This augmented portion of the program

consists of an ISI plan patterned after the ASME Boiler and Pressure Vessel Code, Section XI requirements to the extent applicable to SRS and adapted to the SRS PWS design.

DPSTM-88-100-1 is the initial documentation of a comprehensive ISI program for SRS. The program is designed to completely detail the scope, extent, frequency, and method of examination for each specific component in the PWS for K-, P- and L-Reactors. The program includes implementation details and record-reporting format and will be composed of three volumes. The first volume of the ISI plan serves as a code governing the inspection of the PWS and categorizes the PWS components within the scope of the ISI program.

The ISI plan includes the following reactor pressure boundary items and their supports of the primary water system: (1) reactor tank, including the outlet nozzles; (2) primary coolant piping flanges on piping and components; (3) primary side of the heat exchangers; (4) expansion joints; (5) anchors, supports, and hangers; (6) plenum and plenum inlet nozzles; (7) Bingham pumps; and (8) valves within the PWS, including the first isolation valves, but excluding the heavy water purification and storage system.

A true ASME Section XI type inspection baseline does not exist for the SRS reactors due to the vintage of the plant and other limitations. Therefore, the data from each scheduled inspection will be analyzed to form an equivalent baseline. Examination methods and SRS categorization of weldments based on intergranular stress corrosion cracking (IGSCC) susceptibility are described in detail in Section III of the ISI plan. Inspection schedules, examination methods, frequency, and successive intervals for piping, heat exchangers, and reactor welds are described in the tables of the plan. (The evaluation of reactor tank UT examinations, which are required for restart, is discussed below.

WSRC reported a significant crack-like indication in one of the K-Reactor 16-inch-diameter piping welds in the PWS in early August 1990. The indication was found as a result of the piping reinspection (ultrasonic testing) program. The subject weld had been inspected in 1984 and 1986, and the indication was classified as a weld defect at those times. It was decided to remove the section of the piping containing the indication and conduct metallurgical evaluations and destructive testing to better characterize the defect. The cross-sectional metallographic examination revealed an unusual weld combination of two welds overlapping with a base metal piece inserted at the root; no cracks were found. The change of microstructure at the interface of the weld and the inserted base metal piece may have acted as a reflector of the UT signals. (See Reference 13 for additional details.)

Other Restart Issues

Two reports, "K-Reactor Primary Coolant Systems Integrity" (Reference 14), and "K-Reactor Tank Integrity Status" (Reference 7) were issued in 1989. These reports document that significant evidence exists that the structural integrity of the K-Reactor tank and the K-Reactor PWS is adequate for safe operation. This evidence was based on the results of visual examination; an

evaluation of the design, construction, operating conditions and history; and on an analysis of the potential for service degradation of the materials.

The reports concluded that visual inspections of the K-Reactor tank have shown no unusual indications. The report also indicated that the operating conditions do not adversely affect tank integrity and the material remains stable under normal load and credible accident conditions. These reports have been accepted by DOE; however, recently WSRC reported that another report WSRC-RP-90-099, "Reactor Material Program-Leak Detection System Reliability" (Reference 15) will replace RP-SI-89-02. This item is a part of the ROMP DB-6.4, and has been reviewed and accepted in SER Section 6.7, System Integrity.

An inspection and surveillance program to ensure the integrity of the cooling water system (CWS) piping was implemented as WISR-38 (see restart criterion 2). Piping of concern is that which, if it failed, could lead to process water pump flooding and subsequent failure to provide PWS circulation. The inspection of the CWS piping is complete and is addressed in RTM-5001 (Reference 16). DOE staff review of the CWS piping surveillance program has concluded that it adequately resolves this restart issue.

Reactor Tank Ultrasonic Testing (UT) Program

As an additional assurance for the reactor tank integrity and to establish a long-term surveillance program for the tanks, UT inspections of the tank welds have been initiated. UT of 40 percent of the accessible welds of P-Reactor tank was completed in November 1989; UT of 60 percent of the accessible welds of K-Reactor was completed in March 1990. The P-reactor results were acceptable, and no internal surface-originated cracks were reported. This report has been reviewed and accepted by the staff.

The ISI plan for the SRS PWS specifies UT of the reactor tank on a nominal 5-year frequency. The procedure for the evaluation of the reactor tank inspection results is described in EDG-89.47 (Reference 17); a second document, EDG-89.48 (Reference 18), provides the technical basis for the acceptance criteria; and a third document, EDG-89.49 (Reference 19), describes the proposed initial sampling plan. A cover report, WSRC-RP-89-208, summarizes each of these reports and provides the acceptance criteria for SRS reactor tank UT examinations.

The proposed initial sampling plan provides for inspection of the heat-affected zones (HAZ) of the circumferential and longitudinal welds, portions of base metal, and any repair areas. The plan includes flaw acceptance criteria that are sufficiently conservative to assure tank integrity during both normal operation and postulated accident condition.

The objective of the reactor tank ultrasonic inspection program is to detect IGSCC of HAZ of the tank weldments. The original welds were 100% radiographed during fabrication and installation. Initially, a UT scope of 40% of all accessible welds was adopted for all SRS reactors to sample all critical weld areas. This would include 100% of all main vertical welds and about 33% of the tank circumferential and nozzle attachment welds.

For the P-Reactor, a 40% inspection was conducted and no cracks were identified. After the initial 40% of K-reactor welds were inspected, additional time was available within the allotted schedule and, at management direction, additional UT was completed. This allowed a total of 60% of welds and heat-affected zones to be inspected. No indication of cracking was found.

The UT scope is adequate for restart based on the following justification.

- o An in-depth review of the available stress analyses, design documents, and integrity information has been performed, and the results demonstrate adequate structural integrity, based on the following:
 - The code design margins are satisfied.
 - There was no prior evidence of cracking in earlier visual examinations.
 - The sensitive leak detection system shows no evidence of leakage in the past.
 - Fracture analysis shows that postulated large flaws would remain stable.
 - Experimental data indicate that crack growth rates would be very low under SRS reactor conditions.
- o The 60-percent sampling basis includes the high priority welds, based on their relative fluence accumulation, limiting instability flaw size, highest applied stresses, residual stresses, and fabrication history (Reference 27).
- o The inspection scope and frequency compares favorably with and, in some instances, exceeds commercial reactor practices.
- o The DOE Independent Review Committee (DIRC) for system integrity has reviewed and endorsed the sampling plan. DIRC's conclusion reflects the generic nature of IGSCC in weld HAZ as opposed to an isolated, random form of degradation. Therefore, it may be observed that, since no cracks have yet been found in the large sample inspected, it is unlikely that cracks would be found in the balance of the welds.
- o The adequacy of the 60-percent sampling has also been confirmed by a WSRC statistical analysis.

In addition to the review of general system integrity of the SRS reactors by various review panels, detailed aspects of the UT procedures, inspection work, and examination data have also been reviewed by independent NDE experts. Two recognized experts in the field of nuclear vessel NDE provided independent oversight and conducted random observations of the activities of the UT operation. Both UT and eddy current testing (ET) methods were used for ISI of

the reactor tanks. Additionally, the tank UT inspection personnel and equipment were independently qualified by the Electric Power Research Institute (EPRI) NDE Center. UT readiness reviews were performed by both DOE and WSRC teams prior to the K-, P- and L-Reactor tank UT inspections, and performance qualification reviews for the "Lessons Learned" were initiated after the completion of the UT inspections for P and K reactors. The results of the quality reviews have been applied for the L-Reactor tank UT.

The reactor vessel inspection program consisted of (1) robotic scanning device, (2) Intraspect/98 (I/98) Automated Ultrasonic Imaging System, (3) UT and ET methodologies, and (4) UT and ET examination procedures.

The SRS reactor vessel weld areas are inaccessible for UT and ET from the outside surfaces of the vessels, and access for examination of the weld HAZ areas of concern can only be gained from the tank inside surfaces. WSRC developed a robotic device that is capable of performing remote inspections when inserted into the reactor tank through the 4 3/8-inch-diameter universal sleeve housings (USH) located on the upper plenum. The robot can reach most of the tank welds and can attain 360 degrees coverage when placed in various USH locations.

The Amdata I/98 Automated Ultrasonic Imaging System was coupled to the remote inspection robot and provided computer-based control of the examination process, data acquisition, probe positioning, automated scanning and other information. Data are first stored and later analyzed in a data analysis station. The recorded data include transducer positions, reflector sizing, depth, and associated wave forms. The system also includes high-resolution video cameras and recording equipment and lighting equipment to allow visual observation and manipulator positioning capability. The visual capability also allows examination of test surfaces to confirm flaws on the inside diameter (ID) surfaces.

The ET test system was used to find the weld locations and to scan the circumferential and vertical welds and the HAZ areas to detect IGSCC and in the 3/16-inch-thick expansion ring assemblies at the top of the SRS reactor tanks. Additionally, an ET system is used to confirm the absence or presence of ID surface conditions or reflectors detected in the UT scanning procedure.

The UT and ET equipment and methodologies were reviewed by DOE and their advisory personnel and judged to be adequate to detect IGSCC on the interior wall of the tank, as well as other relevant defects and indications.

Inspection Procedures

Procedures RTIP 001 through RTIP 016 were used. The basic procedures were developed initially by WSRC and qualified by EPRI using the prototype robot, cracked (IGSCC) test plates, and a reactor mock up tank.

UT Acceptance Criteria

Reference 3 describes a three-stage standard based on flaw length. A flaw with an effective length of 10 inches or more exceeds the acceptance standard

and requires further examination and analysis for disposition. Next, any flaw with effective length equal to or exceeding 5 inches (re-examination standard), but less than 10 inches (acceptance standard), is acceptable for continued operation; however, the flaw must be re-examined during the next long shutdown. A flaw whose effective length equals or exceeds 2.5 inches, but is less than 5 inches (re-examination standard), is acceptable for continued operation.

Calibrations

Calibrations were established per RTIP-008, Revision 2 for UT and RTIP-009, Revision 1 for ET. Data acquisition points were set at 0.1 inch increments along the scan path.

Examination Results

Six linear indications were detected in the UT and ET scanning of the welds and the HAZ of the K-Reactor vessel, but supplemental ET and visual examinations showed them to be embedded reflectors (flaws within the weld metal) or weld geometry. These conditions were acceptable per WSRP-RP-89-208, the reactor tank UT acceptance criteria. In addition to the above-mentioned flaws, indications attributed to attachment welds, assembly clips, attachment removal areas were detected on the inside diameter (ID) and outside diameter (OD) base metal areas. The ID surface indications were not cracks, as was confirmed by additional ET and VT scanning.

Eddy Current Test Anomaly

A higher level of indication of magnetic permeability, in a varying degree, was noted in the K-Reactor KH-4 circumferential/horizontal weld when compared to other welds in the K-tank and P-tank. The ISI found no evidence of IGSCC or any crack-like evidence in the KH-4 weld HAZ areas examined nor in any of the weld HAZ areas examined in this K-tank ISI program. It has been suggested that an incorrect filler wire may have been used during fabrication. A broad investigative program was initiated to establish the cause of the permeability difference in the KH-4 weld and to determine whether the anomaly was indicative of an unacceptable condition. Various test samples were fabricated with different filler materials and combinations thereof, using varying weld parameters, to attempt to duplicate the permeability noted in KH-4 weld. WSRC published the results of this investigation in a summary report (Reference 20). The conclusion drawn from the above document is that the welds were made of SS filler material, but a type which yields a higher ferrite content in the finished weld.

Based on these findings the staff determined that both the UT inspection for the K- and P-reactor tank and the overall UT program for SRS have been well organized and the results are satisfactory. WSRC found no evidence of IGSCC or any cracks in the weld HAZ areas examined in the K- and P-reactor ISI program. The discontinuities found in the examinations resulted from manufacturing conditions. All the UT acceptance criteria, UT procedures, and results published to-date have been approved and accepted by DOE.

14.3.4 In-Service Testing (IST) of Pumps and Valves

Periodic equipment tests are currently implemented in a selective and prioritized program based on the importance of valves, pumps, or other equipment to safe operation and shutdown requirements and on analyses of the maintenance history of the equipment. Equipment with records of frequent breakdown and operational trouble were assigned a higher frequency of testing. The schedule and record of these tests and replacements are specified in "Periodic Test and Replacement - Schedule and Records," DPSOL 105-1851A-PLK (Reference 21). Like DPSOL 105-1851B, this document is an operating log that requires sign-off after verification of compliance. Tables 1 through 7 of DPSOL-1851A-PLK dictate nominal test and inspection frequencies of 3 months, 6 months, 1 year, 2 years, 3 years, 4 years, and 5 years. For extended or special reactor outages the test and replacement interval may be exceeded, if specified requirements are met. Included in the inspection are an Emergency Cooling System (ECS) and Emergency Cooling Water (ECW) pump operability check, moderator recovery system (MRS) operation and testing, a functional check of hydraulic valve systems, and cooling water valve checks outside Building 105.

As with DPSOL 105-1851B, DPSOL-1851A-PLK is not a complete and comprehensive program. The effectiveness of the format, the reporting and evaluating method, and the handling of the nonconformances is questionable. Conflict and redundancy may also exist with programs currently under development by WSRC. However, this program is currently supplemented by additional tests specified in the Restart I&T plan and related Reactor Test and Inspection Reports (RTIRs). The staff determined that DPSOL 1851A, as supplemented by this additional testing, form an acceptable basis for restart. All pump and valve test plans will be upgraded as a single plan under post-restart program RSIP-A094 (Reference 22).

Comparison of ASME Section XI Requirements

WSRC has performed a comparison of the existing IST program to the requirements of the ASME Section XI and included it in the restart I&T plan. This comparison meets the restart criteria. Modifications and upgrades are necessary and will be provided through the post-restart RSIP-A094 program.

Motor-Operated Valve Analysis Testing Systems (MOVATS)

This program, which is industry recognized and accepted, is currently being implemented. The schedule target is to test a predetermined number of valves prior to restart. The list of the critical valve operators to be tested has been completed as WSRC documents for ROMP Item SE 14.2. The reports that cover the list are identified as Reactor Technical Memoranda, RTM-4860 and RTM-5047.

WISR-6, "Pump Test and Monitoring for Degradation"

To supplement or upgrade the tests discussed above, WSRC implemented WISR-6, "Pump Test and Monitoring for Degradation." Under this program the ECS, CWS, and sump pumps were tested and/or monitored for flow degradation.

The pump degradation testing requirements are described in RTM-5005 (Reference 23). Pumps tested are listed in the pump tables of section IV of the I&T plan.

WISR-8, Supplemental Valve Inspection and Evaluation Program

WISR-8 has been implemented at SRS. A list of key valves has been prepared based on their importance to safety. These valves were disassembled, and their internal parts were checked and evaluated. The WISR 8 inspection requirements are documented in RTM-4987 (Reference 24). The valves inspected are listed in Section III of the WISR items and in the Section IV valve tables of the restart I&T plan.

14.3.5 Qualification NDE and Test Personnel

WSRC is implementing a qualification program for NDE and test personnel at SRS in conformance to NQA-1. This program is identified in WSRC QA Manual, 1Q (dated April 1, 1990), and is mandated by DOE Order 5700.6. Qualification of NDE personnel is also in accordance with the practices of SNT-TC-1A and Section XI of the ASME Code, as applicable to the SRS reactors.

Qualification and certification of NDE personnel is addressed in Section 9.0 of the WSRC QA Plan. The QA Manual (1Q), Procedure QAP-11, addresses the requirement of "testing personnel." For the reactor tank UT, use of a Level III inspector is required for all tank testing and has been verified from the K-UT test report (Reference 8). UT inspection personnel are qualified to ASME Code Case N-409-1 or pass a performance demonstration as required by the EPRI NDE Center IGSCC program.

14.3.6 ISI and IST Long Term Programs

WSRC is currently developing an ISI and IST plan comparable to commercial nuclear plant practice and in conformance to ASME Code Section XI (to the extent practicable for SRS reactors). Volume 1 of the augmented ISI Plan for the SRS Process Water System is being implemented. Volumes 2 and 3 of the ISI plan are under preparation and shall become effective before the first extended outage after restart. In the same period, the existing programs (DPSOL 105-1851A/B-PLK) and the augmented ISI plan (Volumes 1, 2, and 3) shall be combined to create one comprehensive inspection and testing plan. The comparison of the post-restart I&T Plan with ASME Section XI has been provided in the restart I&T Plan and is satisfactory.

14.4 Conclusion

A number of independent, nationally recognized technical experts have been involved in the review processes for the SRS primary system integrity as well as in the development of the reactor tank UT acceptance criteria. In the process of conducting this evaluation, the staff has been assisted by a panel of experts (DOE Independent Review Committee) in the areas of metallurgy, fracture mechanics, corrosion, and NDE.

The panel conducted an extensive and thorough review of WSRC system integrity and inspection/testing programs, and published the final report on July 26, 1990 (Reference 25). The major conclusions drawn from the Independent Review Panel included the following:

- o Given the design, materials of construction, leak detection, and in-service inspection (ISI) program, the sudden ("instantaneous") formation of large breaks in the pressure boundary is so unlikely as to be almost incredible, for any foreseen operating conditions.
- o A very conservative program is being followed to set up the ISI plan.

Based on the above evaluation, the staff concludes that all restart criteria have been satisfied.

14.5 References

1. WSRC, WSRC-RP-89-368, Vols. 1 - 4, "Reactor Operations Management Plan (ROMP)," Rev. 2, August 31, 1989.
2. Westinghouse, "Savannah River Plant K-Reactor Restart Strategy" (Charlotte Criteria), November 25, 1988.
3. WSRC, WSRC-RP-89-208, "Reactor Tank Ultrasonic Acceptance Criteria," May 1989.
4. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section XI, 1986 Edition with 1987 Addendum.
5. U.S. Department of Energy (DOE), SR 5700.6C, "Savannah River Operations Office (SR) Order: Quality Assurance," February 2, 1987.
6. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan," (Applicable Sections).
7. WSRC, WSRC-RP-89-152, "K-Reactor Tank Integrity Status," May 1989.
8. WSRC, WSRC-TR-90-42-123, "K-Reactor Restart Inspection and Testing Plan," Rev. 0, October 17, 1990.
9. WSRC, DPSOL 105-1851B-PLK, "Periodic Equipment Inspection - Schedule," Rev. 14.
10. WSRC, "Quality Assurance Manual 1Q," April 1, 1990.
11. SNT-TC-1A-1984, "Recommended Practice for Nondestructive Test Personnel Qualification and Certification."
12. WSRC, DPSTM-88-100-1, "ISI Plan for the Savannah River Production Reactors Process Water System," December 1988.

13. WSRC, WSRC-TR-90-504, "Metallurgical Examination of Weld 11 from Process Water Pipe 1K-PW1-16," December 19, 1990.
14. WSRC, RP-SI-89-02, "Primary Coolant System Integrity - K-Reactor Restart Program," May 1989.
15. WSRC, WSRC-RP-90-099, "Reactor Material Program Leak Detection System Reliability," February 28, 1990.
16. WSRC, RTM-5001, "Cooling Water System Inspection Review, K-Area," Rev. 1, June 25, 1990.
17. WSRC, EDG-89.47, "Procedure for Evaluation of Reactor Tank Inspection Results."
18. WSRC, EDG-89.48, "Technical Basis for The Savannah River Reactor Tank Acceptance Criteria."
19. WSRC, EDG-89.49, "Reactor Tank UT Program - Sampling Plan."
20. Letter, OPS-RRD-900133, from Gallagher to McCoy, "Investigation of Eddy Current Signals and Structural Integrity of Horizontal Weld 4 in K-Reactor Tank," dated October 26, 1990.
21. WSRC, DPSOL 105-1851A-PLK, "Periodic Test and Replacement - Schedule and Records," Rev. 14.
22. WSRC, WSRC-RP-89-379, "Development and Management of Reactor Safety Improvement Program," July 1989.
23. WSRC, RTM-5005, "WISR-6 Pump Degradation K-Area," September 15, 1989.
24. WSRC, RTM-4987, "Critical Valve Inspections," October 4, 1989.
25. Shewmon, Doctor, Wilkowski and Zahoor, DOE Independent Review Committee's Final Report, "Savannah River Reactors Primary System Integrity," July 26, 1990.
26. WSRC-1-05, Rev. 1, "Quality Assurance Plan", October 31, 1990.
27. WSRC-SRL-MAT-910032, Rev. 1, "Scope of K-Reactor Tank Ultrasonic/Eddy Current Inspection: Adequacy for Restart-Response to Defense Nuclear Facilities Safety Board Inquiry", March 20, 1991.

15 EMERGENCY PREPAREDNESS

15.1 Background

In 1988, the Department of Energy (DOE) performed an appraisal of the Westinghouse Savannah River Company (WSRC) Emergency Preparedness (EP) Program. A comparison of the EP Program for the Savannah River Site (SRS) with current commercial nuclear industry standards revealed a number of areas that required improvement. This appraisal focused on compliance with the listed restart criteria, as well as the DOE Orders listed. Although the team found several positive attributes of the program, the team determined that the Emergency Preparedness Program had deficiencies.

Although improvements have been made in the EP program for SRS, the new DOE safety culture and concern for the health and safety of the public have dictated that additional improvements be made prior to the restart of K-Reactor. As a result, the restart criteria outlined below have been developed to ensure that an effective EP Program is in place prior to restart. The implementation of activities to meet these criteria will provide a basis for protecting the health and safety of both onsite and offsite personnel in the unlikely event of a reactor accident.

A comparison among the restart criteria, the DOE 5500 Series Orders, and the SRS Emergency Plan (WSRC 6Q) (Reference 1) was undertaken. This comparison demonstrated that the SRS Emergency Plan, together with the appropriate procedures (i.e., MRP 4.08 [Reference 2] and EPIP 200 [Reference 3]) and the restart acceptance criteria satisfy all of the requirements contained in these Orders. A report documenting this evaluation has been prepared and is on file.

15.2 Restart Criteria

The criteria for determining the adequacy of the SRS Emergency Preparedness Program for restart are described below.

(1) General

- (a) An emergency preparedness program shall be developed and implemented for coping with emergencies. The 100-K program shall interface with emergency plans implemented for the overall SR Site and be an integral part of the site emergency plans. The program shall include (1) a description of the response organization established for responding to an emergency situation that may arise, (2) the plans and procedures to be followed, (3) the staffing and training needs of the emergency response organization, (4) the facilities, equipment and resource requirements, (5) emergency assessment and notification procedures, (6) personnel protection procedures, (7) criteria for how the effectiveness and readiness of emergency operations will be determined, and (8) interfaces with the overall SR Site Program.

(b) The emergency response organization for the SRS reactors shall be clearly documented by the Contractor, and as a minimum, include people experienced in the following areas:

- o Reactor Operations
- o Operations Support Center (OSC) operations
- o Medical/first aid
- o Search and rescue
- o Fire brigade
- o Radiation protection
- o Security
- o Area emergency coordination operations

(c) The reactor emergency response organization (ERO) shall identify, by emergency response position, candidates for the primary and backup response team staff who possess the requisite knowledge, skills, and qualifications for the positions occupied. The responsibility and authority of each ERO manager, supervisor, and team member shall be clearly defined, either in the emergency response plan (discussed below) or in position-related and specific procedures. The chain of command and communication link at the reactor sites and with SR Site Technical Support Center (TSC)/Emergency Operations Facility (EOF) shall be clearly documented.

(2) Reactor Emergency Plan

(a) The contractor shall implement an emergency response plan which incorporates the emergency preparedness program features including:

- o The emergency response organization, authorities, responsibilities and relationship with the SR Site TSC/EOF, with a defined line of succession established for the emergency coordinator position, and conditions for a higher level official assuming that function; and the non-delegated responsibilities for emergency classification defined.
- o A description of the emergency classification system and associated emergency action levels (EALs) consistent with DOE Order 5500.2A, and reporting requirements for notifying the TSC/EOF of potential/actual emergencies.
- o Provisions for notification and activation of the reactor emergency response team, including the primary and backup means of notification and communication.
- o Identification of operations of credible emergencies including the technical assessment and mitigation of reactor conditions, radiological conditions, and core damage.

- o Reactor emergency response team coordination and interface with the TSC/EOF, including the transmission of information on the nature and magnitude of hazardous material releases into and from the reactor buildings, as well as a description of intra-area information flow including periodic briefings of the central control room (CCR), OSC, and rescue teams by the area emergency coordinator.
- o Personnel protection procedures including evacuation, assembly and accountability of the reactor protective and limited areas.
- o The organization, authority and responsibility for implementing reentry and recovery actions including first aid, search and rescue, and emergency repairs.
- o A format, content, and implementing procedures consistent with the SR Site Emergency Plan.

(3) Staffing and Training

- (a) The reactor emergency response organization shall possess the requisite knowledge and practical abilities obtained from day-to-day on-the-job experience and/or through a documented training program in the following areas:
 - o SR Site and 100-Area Emergency Plans and implementing procedures
 - o reactor emergency facilities, equipment, and resources
 - o reactor communications and information flow process
 - o reactor-specific job knowledge and skills, including applicable precautions and limitations
 - o command, control and communication responsibilities in the reactor control room, TSC, and EOF
- (b) The training received by reactor emergency response team members shall be documented and a mechanism put in place to ensure that all required training is satisfactorily completed prior to the assignment of the individuals in the emergency response organization. A mechanism shall also be implemented to ensure that continuing training is provided to emergency response team members at least once a year. Additionally, all personnel employed at the reactor sites shall have completed training in evacuation, assembly, and accountability. Documentation of that training shall be established and maintained. Finally, an orientation briefing covering the minimum knowledge on what to do

in an emergency with respect to evacuation, assembly, and accountability shall be given to all unescorted visitors who enter the reactor areas, prior to, during, or after reactor restart.

(4) Emergency Facilities, Equipment, and Resources

- (a) An Operations Support Center (OSC) sized, located, and equipped consistent with its emergency function, to support emergency responses shall be established. In addition, there shall be a pre-established policy governing situations where formal job requirements may be bypassed during an emergency situation provided that safety will not be compromised. REMACS or REDACS information shall be available to the TSC. Radiation monitors relied upon for emergency detection, classification, and assessments, shall be in place and operable.
- (b) An implementation plan shall be developed, submitted, and approved to ensure the reactor Public Address Alarm System is used to provide complete coverage of an emergency situation and audibly communicate information in all normally occupied areas of the site. The reactor control room shall have direct control of the Public Address Alarm System from which emergency signals can be initiated directly and from which other users of the paging system can be overridden. The reactor area rescue and reentry teams shall be provided with a means to communicate with the area emergency coordinator. Available methods of communicating shall be specified in the reentry procedures and the method most suitable for the situation shall be used.
- (c) Up-to-date copies of emergency preparedness and engineering documentation needed to support the mitigation of reactor area emergencies shall be readily available to the OSC, the TSC, and the EOF.
- (d) Sufficient emergency equipment inventories for rescue, first aid, and radiological monitoring shall be available before reactor restart and be maintained in appropriate quantities subsequent to reactor restart. In addition, sufficient and reliable primary and backup communications networks, including emergency intra-area communications shall be available to accommodate normal operations and emergency needs.

(5) Emergency Assessment and Notification Procedures

- (a) The operating contractor shall implement formal procedures for notifying onsite reactor areas of emergency conditions, including those areas which may not currently be serviced by SR sitewide public address systems. In accomplishing this, the contractor shall conduct an extensive in-house interdisciplinary review and validation of all reactor EAL indicators contained in DPSOL 100-4000 and revise any and all related procedures as necessary to address review comments and lessons learned from past emergency

exercises. Mechanisms shall be established to ensure that the technical basis for EAL definitions are documented, and DOE approval obtained prior to changing existing EALs or implementing newly developed EALs.

In addition, procedures shall be implemented for assessing releases of radiological and non-radiological hazardous materials, including methods for measuring contamination levels at the SR reactor facilities, onsite and offsite, and for determining the magnitude of the source terms and projected doses that may be released.

- (b) The emergency procedures and related training for the reactor emergency coordinators and emergency duty officers at the reactor sites shall collectively cover the requirements for fast moving events of an immediate general emergency, for prompt accident classification, notification of the TSC, and for the transmission of offsite Protective Action Recommendations (PARs) over the Emergency Notification Network (ENN) within 15 minutes of the classification of an emergency, or recognition that a general emergency condition exists. Default PARs, based on radiological releases (doses) and/or plant conditions shall be included in appropriate protective action decision-making procedures.

(6) Personnel Protection Procedures

- (a) Procedures shall be implemented which specify allowable personnel radiation exposure limits, and which will ensure that radiation exposures can be accurately determined and recorded. Individual radiation exposure limits shall be established for emergencies such as saving a life or protecting vital equipment with procedures to designate who is authorized to permit those limits to be exceeded when emergency operations are involved.
- (b) Emergency procedures shall be provided for the decontamination of personnel, equipment and facilities, and the habitability of essential personnel in the reactor area emergency response facilities. Dedicated decontamination supplies and facilities shall be readily available and identifiable. Respiratory equipment and supplies shall also be readily available and identifiable and the quantities of such supplies (air bottles, air-breathing apparatus, etc.) appropriately established and documented. A method for recharging the bottles shall be available.
- (c) A system for personnel accountability shall be put in place which shall contain the following provisions:
 - o Accountability procedures, rally points, and reactor area emergency notification procedures to accommodate the unique hazards posed by the SRS reactors (e.g., filter compartment radiation, uncontained noble gas releases, etc.) and ensure

that all personnel working in the reactor protected areas, as well as visitors and temporary personnel, can be accounted for within 30 minutes of the emergency notification.

- o The conduct of evacuation drills to verify the adequacy of accountability procedures.
 - o Developed formal procedural guidance for protective action requirements for essential and non-essential reactor area workers.
- (d) Pre-arranged plans, training, and agreements for fire fighting support shall be established by the contractor, as well as provisions and procedures for facility or site evacuation with evacuation routes clearly marked and kept free of obstacles. Personnel monitoring equipment and procedures shall be in place for the protection of security force personnel who may be required to remain at their post during a radiological release emergency.

(7) Emergency Drills and Exercises

Emergency drills and exercises shall be conducted prior to restart, and at least annually thereafter, to test and verify the adequacy of the reactor emergency plans. The drills and exercises shall (1) Demonstrate the effectiveness of coordination between emergency response team members in the reactor control room, the TSC and the EOF; (2) Demonstrate the effectiveness of the command and control of the reactor during the assessment, mitigation, and recovery phase of a simulated accident at the reactor areas; (3) Demonstrate the ability of the emergency response team to carry out shift duties until they are augmented with TSC support staff; and (4) Demonstrate the adequacy of TSC/EOF support of the reactor area in dealing with organizations external to the DOE/SR, its Operating Contractor, and its Security Contractor.

(8) Sitewide Emergency Program

- (a) The reactor area emergency plan shall be an integral part of the SRS Sitewide emergency program. The program shall require DOE approval. The SRS site emergency plans shall effectively incorporate the following program features:
 - o Training of offsite protective action decision makers, including the relationship between reactor conditions and protective measures.
 - o Coordination and communication between TSC/EOF and the reactor area Emergency Response Organization and emergency response facilities.

- o Procedures and resources for transportation of evacuees from designated area assembly and accountability rallying points to safe areas.
 - o Agreements and provisions for providing transportation and medical treatment of contaminated and injured personnel.
 - o Non-delegatable responsibilities for emergency classification, Protective Action Requirements (PARs) and offsite notification established and consistent with the line of succession for the emergency coordinators position.
 - o Personnel with reactor safety analysis expertise for augmenting reactor emergency response teams.
 - o Formal training on emergency dose assessments implemented to ensure Weather Information and Display System (WINDS) operators can rapidly and accurately assess the potential consequences of a radiological release.
 - o EOF responsibilities and authority specified between DOE/SR, DOE/SRSP0, and DOE Contractors.
- (b) Page-Comm type radio receivers, or an acceptable alternative shall be provided for emergency response team members, along with adequate instructions to remote groups expected to be within 2.5 miles of the reactor areas, the operation of which shall be verified quarterly, as well as during emergency exercises and drills.

15.3 Evaluation

General

The staff reviewed the SRS Emergency Plan (WSRC 6Q), the RAEP (WSRC 6Q7) (Reference 4), and corresponding response EIPs (References 5 - 16), and concluded the following with respect to the Reactor Area ERO.

Only four supervisory ERO positions are identified in the RAEP:

- o Area Emergency Coordinator (AEC)
- o Operations Support Center (OSC) Coordinator
- o Health Protection (HP) Coordinator
- o Protective Force Supervisor

All other non-supervisory ERO positions are addressed as CCR Staff, OSC Staff, and Security. The response EIPs identify other potential ERO positions (e.g., Accountability Coordinator, Response Team Leader, Rally Point Coordinator, and HP Inspector), which are assigned from the OSC Staff. Administrative EIP 6Q7-302 identifies Reactor Area Organization administrative position titles and their correlation to ERO supervisory and staff positions. EIP 6Q7-302 also identifies the EP training requirements

for the ERO positions, but indicates that fire response training, respiratory protection qualification, and medical/first aid response training are controlled and documented by organizations other than EP. Although the administrative position titles and training listed imply that the Reactor Area ERO includes personnel experienced in the areas listed in the restart criterion, documentation showing the job-specific qualification requirements necessary for assignment to the ERO and the administrative position titles were not available. For this reason, the requisite knowledge, skills, and qualification necessary for assignment to the ERO could not be fully determined. This is an open item.

The four specific supervisory ERO positions and three separate non-supervisory staffs have been identified as coming from the shift crews, with relief personnel supplied by the oncoming shift crews; however, the basic number of each type of personnel (e.g., operations, HP, maintenance, power, security, etc.) involved has not been clearly defined. Additionally, information on a 1-hour augment to the shift staff (similar to commercial industry practices) and the onsite or offsite locations from which they are obtained have not been addressed. For these reasons, candidates for primary and backup response team staff could not be identified, as required by the restart criteria. This is an open item.

The responsibilities and authority of each Reactor Area ERO management and supervisory position have been clearly defined in the RAEP and the response EIPs. The responsibilities of the ERO, OSC, and CCR non-supervisory staffs are addressed in the response EIPs; the security staff responsibilities are addressed in security procedures.

The chain of command between the Reactor Area and the Technical Support Center(TSC)/Emergency Operations Facility (EOF) is clearly defined; however, the communications links between the area and site facilities are not clearly addressed. Communicator ERO positions have not been identified, and specific information requirements have not been addressed. The plans and procedures indicate that all Reactor Area and Reactor Area to Site emergency facility communications are performed by the supervisory ERO members. These are open items.

Closure of this restart criterion is dependent upon correction of the open items listed above and on formal approval of the SRS Emergency Plan and EPIP 6Q7-001. Revisions made to the referenced plans and procedures should be reviewed to ensure continuing compliance with the restart criterion.

Reactor Emergency Plan

The staff reviewed the RAEP and response EIPs to evaluate the Reactor Area ERO authorities and responsibilities and their relationship to the site emergency response mechanisms. This review included the lines of succession for the emergency coordinator position, conditions for a higher-level official assuming the function, and the non-delegable responsibility for emergency classification.

The staff determined that the authorities, responsibilities, and relationships of the supervisory Reactor Area ERO positions to the site organization, addressed by the RAEP, have been defined. The line of succession for the emergency coordinator position at the reactor area has been clearly defined, both to a higher-level official and to the next in line in the CCR.

The line of succession for the emergency coordinator position to the site level is not addressed directly, but is inferred by the AEC responsibilities and discussions on the concepts of operations within the plan. In addition, emergency classification is clearly addressed, but there is no mention of it being a non-delegable responsibility. These are open items.

The staff determined that the remaining ERO personnel are grouped into three staffs (CCR Staff, OSC Staff, and Security) and do not appear to have any specific authorities, responsibilities, or relationships to the organization at the site emergency facilities.

The staff ascertained that the RAEP and response EPIP 6Q7-001, "Emergency Classification," describe the reactor area Emergency Action Levels (EALs) and classification system consistent with DOE Order 5500.2A.

The staff reviewed the RAEP and EPIP 6Q7-002, "CCR Staff Actions for Declared Emergencies," and determined that these documents adequately address reporting requirements for notifying the TSC of potential/actual emergencies.

The staff reviewed the SRS Emergency Plan, RAEP, and response EIPs, and noted that the CCR will notify and activate the area on-shift personnel with a Public Address (PA) announcement, while the TSC retains the responsibility for notification and activation of the response team from external locations. The RAEP also indicated that the backup for the CCR announcement was a duplicate PA system capability from the Security Data Assessment Console area. Additionally, conventional telephones provide supplemental backup capability.

From a review of the SRS Emergency Plan, the staff established that the TSC has primary and backup capabilities for onsite and offsite notifications and communications. The TSC has a pager system for notification of ERO personnel and the SS-1 for notification of other site facilities of personnel requirements. Conventional telephones provide backup notification and communication capability.

The staff reviewed and evaluated the sections of the SRS Emergency Plan, RAEP and associated EIPs, and Reactor Areas Abnormal Conditions Control Procedures (100-700 Series) (Reference 17) dealing with identification of operations of credible emergencies, technical assessment and mitigation of reactor conditions, radiological conditions, and core damage.

The staff determined that the SRS Emergency Plan and RAEP do not contain information on credible emergencies (either operational or safeguards and security), nor do they contain information on technical assessment and mitigation of reactor conditions, radiological conditions, and core damage. However, these documents do reference the Safety Analysis Report (Reference 18) as a source of this information. The Reactor Area Abnormal Conditions

Control Procedures (100-700 Series) and EIPs address actions and groups to perform technical assessment and mitigation of reactor conditions, radiological conditions, and core damage.

The RAEP and response EIPs have been reviewed and evaluated by the staff for adequacy of discussion on the coordination, interface, and transmission of information between the Reactor Area and the TSC and EOF. The staff determined that the discussion on the transmission of information about the nature and magnitude of hazardous material releases into and out of the reactor building, as well as a description of intra-area information flow (including periodic briefing of the CCR, OSC, and rescue teams by the AEC) is unclear and incomplete. This is an open item.

The staff determined that the RAEP and response EIPs give the AEC the responsibility for providing the TSC with information on the status of the emergency, while the HP Coordinator has the responsibility for providing radiological information to the TSC. The RAEP and response EIPs state that the AEC briefs the CCR; the OSC Coordinator and the HP Coordinator brief and debrief the Emergency Response Teams assigned plant entries; and the HP Coordinator discusses radiological considerations with the Dose Assessment Center Coordinator in the TSC and briefs/advises the OSC Coordinator and AEC on radiological conditions and protection requirements. However, the staff has noted a lack of guidance on the type of information that is exchanged. This is an open item.

The applicable Reactor Area EIPs (i.e., EIPs 6Q7-002, -101, -102, -201, -202, -203, and -205) still need to be revised to include guidelines on types of information and frequency of communications regarding the nature and magnitude of hazardous material releases into and out of the reactor building. This includes communications from the reactor area to the TSC, periodic briefings by the AEC to the CCR Staff and OSC Coordinator, and periodic briefings by the OSC Coordinator to the OSC Staff. This is an open item.

The staff reviewed the RAEP and its implementing procedures, which address sheltering, evacuation, assembly, and Protected Area accountability. The staff noted that Protected Area personnel accountability, in the event that crash gates are used and/or the key card reader system fails, is not addressed even though the use of crash gates is specifically addressed in the RAEP. This is an open item.

The staff reviewed the RAEP and various implementing procedures concerning reentry actions (i.e., first-aid, search and rescue, and emergency repairs) and found them adequate. The staff determined that the discussion on recovery actions is also adequate, considering that the EOF has the responsibility for planning the recovery and that the responsibilities of personnel in the reactor area are limited to implementing the directives of the recovery organization within the confines of normal administrative and specially prepared procedures.

The staff reviewed the RAEP and its implementing procedures for consistency with the SRS Emergency Plan. The staff noted that the interface between the documents is adequate to meet the restart criterion. Closure of this criterion

is dependent upon correction of the items listed above and on formal approval of the SRS Emergency Plan and EPIP 6Q7-001. Revisions made to the referenced plans and procedures should be reviewed to ensure continuing compliance with the restart criterion.

Staffing and Training

The staff determined that a documented EP training program has been established for the K-Reactor Area ERO. This program provides training to the reactor area ERO on (1) SR Site and Reactor Area Emergency Plans and implementing procedures; (2) reactor area emergency facilities, equipment, and resources; (3) communications capabilities and information flow processes; and (4) the command, control and communication responsibilities of the Central Control Room (CCR), Operations Support Center (OSC), Technical Support Center (TSC), and the Emergency Operations Facility (EOF).

The staff determined that Section III.K, "Training," of the RAEP, and administrative EPIP 6Q7-302, "Emergency Response Organization and Emergency Training," are adequate with respect to defining the identified Reactor Area ERO position EP Training Program requirements on the following basis.

- o RAEP, Section K.3., "Training," states: "Training for all individuals assigned to the emergency positions will be successfully completed prior to their initial performance in the emergency organization."
- o EPIP 6Q7-302, Attachment 8.1, "Reactor Area ERO Training Matrix", provides training requirement information for each reactor area ERO position and Attachment 8.2, "ERO Training Modules," provides information on the reactor area EP training courses used to train and qualify the Reactor Area ERO. EPIP 6Q7-302 also establishes provisions for documenting training attendance and maintaining applicable EP training records.
- o A training matrix (Reference 19) has been developed that includes the names of personnel required to complete 24-hour staffing of the organization.
- o The RAEP and EPIP 6Q7-302 address provisions for annual EP refresher training to the reactor area ERO.

The staff notes that on May 31, 1990, WSRC instituted a General Employee Training (GET) Program for the reactor area. The 8-hour course provides information on personnel evacuation, assembly, and accountability for the reactor areas. All reactor area employees are required to attend GET and obtain proof of successful completion of the training (i.e., sign-off by the instructor). If they fail the course, reactor area employees are not permitted unescorted access to the protected area. The staff noted that the GET course lesson module, while satisfactory, lacks information on how personnel accountability will be handled.

The staff determined that beginning November 1, 1990, unescorted access was not allowed for badged SRS employees or visitors who have not completed GET. Employees and visitors who have not completed this requirement are escorted by a 100-Area employee who has fulfilled this training requirement.

The staff also determined that emergency preparedness training for the reactor area ERO (i.e., AEC, OSC Coordinator, HP Coordinator, CCR Personnel, and OSC Personnel) has been developed by the WSRC Reactor Training and Procedures (RTAP) group and that RTAP has qualified instructors available to teach the courses. The staff has also ascertained that the 100-Area Training Management Information System (TMIS) will be used by RTAP to document and track emergency preparedness training. This criterion remains an open issue pending formal approval of the SRS Emergency Plan.

Emergency Facilities, Equipment, and Resources

The staff determined that WSRC has designated primary and alternate OSC locations. The Personnel Wing Health Protection Area has been identified as the primary OSC, while the Crane Maintenance Area has been selected as the alternate OSC for K-Reactor. The staff determined that the new primary OSC location appears to be of sufficient size to adequately perform emergency functions.

The staff determined that the amount of communications equipment installed in the OSC appears to be consistent with the OSC functions and the emergency response addressed in the RAEP and in the OSC EIPs 6Q7-101, "Activation and Operations of the OSC," and 6Q-102, "Emergency Response Teams."

The staff reviewed draft EPIP 6Q7-303, "Emergency Equipment and Supplies," and surveyed the emergency equipment lockers located near the OSC to determine if sufficient emergency equipment and supply inventories were available to support OSC emergency functions and responses, including search and rescue, fire support, assessment, repair and modification, and radiological monitoring. The inventories appeared to be adequate except that only 6 high-range dosimeters are stored in emergency equipment lockers in both the OSC and CCR facilities. Considering that there is a potential for more than 6 personnel in each facility during an emergency, this is inadequate. This is an open item.

This EPIP also ensures that the equipment is maintained in a state of readiness by identifying maintenance and inventory requirements. The health protection (i.e., radiological monitoring) instruments are checked in accordance with DPSOL 292-419, "Plant HP Emergency Equipment Audit" (Reference 20). This procedure has not been reviewed by the staff. Maintenance and inventory requirements for first-aid are in accordance with the Medical Department Manual 3Q-7, "First-Aid and Nursing Procedures 4008, Checking Treatment Room and Stocking Emergency Cabinet" (Reference 21). This procedure has not been reviewed by the staff.

EPIP 6Q7-303 also addresses inspection and inventory of rescue equipment and inspection and testing of radio equipment that could be used by the emergency responders. EPIP 6Q7-303 does not address either the inventory of rescue equipment or periodic testing of the OSC telephones to ensure readiness. This is an open item.

EIPs were available in the OSC, and the staff was told that plant equipment documents (e.g., prints, drawings, operations procedures, etc.) could be obtained from the Satellite Document Control Center located in the Reactor Area 105 building. The document control staff is responsible for ensuring the readiness condition of the Satellite Document Control Center. Additionally, the maintenance shops are located near the OSC, where additional equipment, prints, etc., can be obtained for emergency use.

The RAEP and response EIPs adequately describe the emergency response functional requirements of personnel. A pre-established policy to bypass job requirements in emergency conditions, provided that safety is not compromised, is addressed in Standing Instruction No. II-8 (Reference 22). This portion of the restart criterion is satisfied.

The staff observed that a REMACS terminal is located in the TSC. The staff has ascertained that this computer link can only be used for monitoring and enables TSC personnel to read and obtain a hard copy of data screens showing process and radiological parameter status for each of the P-, L-, and K-Reactors. In addition, these data screens provide a complete overview of the reactor status.

The staff noted that the RAEP indicates that REMACS provides remote monitoring of specific system parameters and control functions for critical equipment in the Emergency Power Systems, Process Water Systems, Cooling Water System, Emergency Cooling System, Building Flood Control Systems, Ventilation System, and Radiation Monitoring Systems. The instruments and components monitored provide for the detection of abnormal conditions within the system prior to release, as well as for monitoring the facility and equipment status throughout an accident.

The validation of the in-place and operable condition of the radiation monitoring instrumentation associated with emergency detection, classification, and assessments requires further information and physical confirmation. The staff determined that the radiation monitors that are required for emergency detection are indicated in EPIP 6Q7-001, "Emergency Classification." The specific instruments that are used to detect the accident conditions are identified by name or number. A correlation to the individual radiation monitors will be necessary in order to properly inventory devices during future inspections. This is an open item.

The staff determined that Reactor Engineering has completed an assessment of the K-Reactor Public Address System to ensure functionality during emergency response. The assessment was documented in a letter (OPS-RES-900436, dated November 19, 1990). Even though the communication equipment conforms to that identified in the EIPs, a recent assessment has determined that the PA System has several "dead spots." No mechanism is presently in effect to provide

sufficient and reliable communications to these locations. Accordingly, a compensatory means of communication should be implemented for the "dead spots" until the PA Systems deficiencies are corrected. A system upgrade implementation schedule will be issued upon receiving authorization to perform the tasks. WSRC needs to complete an implementation plan, which details the extent of work required, and obtain DOE approval for implementation. The staff noted that the L-Reactor Area already has the proposed system in place with flashing lights for noisy areas, and has enhanced the alarm coverage. This is an open item.

During the control room visit, the staff observed the presence of the red microphone cut-off switch used to override page users and initiate signals. This switch provides the reactor control room with direct control of the Public Address Alarm System. The staff also witnessed a test of the system, which confirmed its functional adequacy.

The staff reviewed the RAEP and OSC response EIPs 6Q7-101 and 6Q7-102, which identify methods and options of communication to be used by the reactor area response and rescue teams to communicate with the OSC Coordinator. The response EIPs also provide guidance on selecting the most appropriate method of communication. The staff also reviewed administrative EIP 6Q7-303, which adequately describes the method for ensuring portable radio availability and readiness.

The staff determined that provisions to assure the availability of the engineering and emergency preparedness documentation in the Reactor Area emergency facilities are not addressed in the RAEP or the response EIPs. The staff was informed that the responsibility for emergency preparedness document control was given to the Reactor Area Satellite Document Control Center in the 105 Building. The staff was also told that the Control Center contains the most up-to-date engineering documents and is routinely used as the document source. Additionally, the staff was informed that the Satellite Document Control Center is manned on a 24-hour basis and that the Center's personnel would be added to the Reactor Area ERO, which would be addressed in the next revision of the response EIPs. The Document Control Center procedures that ensure controlled distribution of emergency preparedness documents to the appropriate Reactor Area Emergency Response Facilities, as well as ensuring up-to-date engineering documents, have not been evaluated by the staff. WSRC informed the staff that there are plans to establish a technical library to service both the TSC and EOF; however, this effort is presently in its developmental stage. Identification of present provisions for ensuring that up-to-date EP and engineering documents will be available for emergency response to reactor incidents needs to be addressed in the SRS Emergency Plan, the RAEP, and/or appropriate EIPs. The staff needs to evaluate the Document Control Center procedures to assure controlled distribution of the EP documents is satisfactory. This is an open item.

The staff determined that EIP 6Q7-303, "Emergency Equipment and Supplies," identifies the equipment inventory that is stocked in the emergency equipment cabinets and lockers in the reactor area. This procedure identifies maintenance and inventory requirements for the health protection (i.e., radiological monitoring) equipment to be in accordance with DPSOL 292-419,

"Plant HP Emergency Equipment Audit," which has not been reviewed by the staff. In addition, EPIP 6Q7-303 identifies maintenance and inventory requirements for first-aid equipment to be in accordance with the Medical Department Manual 3Q-7, "First-Aid and Nursing Procedure 4008," and "Checking Treatment Room and Stocking Emergency Cabinet," which have not been reviewed by the staff. EPIP 6Q7-303 also addresses inventory and inspection of rescue and radio equipment.

In April 1990, the staff inspected the radios, Bell telephones, Bell Intercom System inter-facility black phones, and the Public Address Alarm System. This equipment accommodates both normal operations and emergency needs. The communication equipment surveyed conforms to that identified in the EIPs.

Closure of this restart criterion is dependent on correction of the items listed above and on the formal approval of EPIP 6Q7-001. Revisions made to the referenced procedure should be reviewed to ensure continuing compliance with the restart criterion.

Emergency Assessment and Notification Procedures

The staff reviewed the RAEP and EPIP 6Q7-002, "CCR Staff Actions for Declared Emergencies," and determined that they adequately address notification by Reactor Area personnel of changing conditions that could affect locations beyond the limited-area fence.

WSRC has issued EPIP 6Q7-001, which corrects some of the deficiencies in DPSOL 100-4000 (Reference 23). WSRC has included refinements to EPIP 6Q7-001 to improve its usability and to include appropriate information based on review comments and lessons learned. However, EPIP 6Q7-001 still needs further modification to satisfy this criterion. This is an open item.

The staff noted that WSRC has issued a Technical Basis Document (TBD) (Reference 24) for the EAL indicators identified in EPIP 6Q7-001. The TBD, which documents the technical basis for the EAL definitions, is presently being evaluated and upgraded to ensure proper validation of the EALs. In addition, discussions are being held between WSRC EP and the Reactor Training and Procedures (RTAP) groups to formally establish a procedure that directly links a change in an EAL to a corresponding TBD revision. This is an open item pending issuance of the procedure.

Even though the SRS Emergency Plan dictates that DOE approval is required for EIPs, the staff was informed that a formal mechanism to ensure that DOE approval is obtained prior to changing existing Reactor Area EALs or implementing newly developed EALs does not exist. DOE approval is being pursued, but not in accordance with a formal procedure. The responsibility for ensuring that DOE is in the routing and review cycle for the Reactor Area Emergency Classification procedure has recently been transferred to the RTAP group. The WSRC EP staff is presently working with RTAP personnel to formally implement this SRS Emergency Plan requirement. This is an open item.

The staff reviewed EPIP 6Q7-001 and determined that measurement of releases of hazardous material are limited to radioactive material releases. The staff was informed that no other significant hazardous material exists on the reactor sites. The staff has also noted that the primary assessment of reactor area hazards involves the measurement of gross gamma radiation by reactor area process radiation monitors, such as the total stack activity monitor and the filter plenum break-through monitor. These monitors are hardwired directly to the REMACS computer and to the Weather Information and Display System (WINDS) computer, which are located in the TSC. In addition, EPIP 6Q7-101, "Activation and Operation of the OSC," and EPIP 6Q7-102, "Emergency Response Teams," address dispatching teams to monitor radiological conditions with portable detection instruments. The EIPs provide adequate directions for ensuring appropriate input to the site ERO in the TSC for the determination of onsite and offsite release consequences. The TSC EIPs direct the use of REMACS and WINDS units to obtain radiological release information, and the Reactor Area EIPs for the CCR and OSC direct lead personnel to provide release parameters and radiological conditions information to the TSC.

The WINDS Users Manual (Reference 25) provides information on the operation of WINDS for source term and dose projection. The staff has not determined the adequacy of the source term and dose projection procedures by review of the approved WINDS Users Manual. The staff needs to verify that there are no significant hazardous materials other than radioactive materials in the Reactor Area. These are confirmatory items pending completion of the staff review.

The staff reviewed the RAEP and EIPs 6Q7-001 and 6Q7-002 and determined that they adequately address the actions and training of the AEC covering the requirements for a fast moving event corresponding to a General Emergency classification. Procedures address General Emergency EALs and classification, notification of the TSC, and protective actions. The SRS Emergency Plan section on training requires the AECs to attend training involving the recognition of general emergency conditions, prompt emergency classification, notification of the TSC, and protective action decisions. Because they are site-level responsibilities, the requirements for procedures that address 15-minute ENN notification and Protective Action Recommendations and training the EDO in the notification protective action recommendation process are discussed under "Sitewide Emergency Program", below.

Closure of this restart criterion is dependent on correction of the items listed above and on formal approval of the SRS Emergency Plan and EPIP 6Q7-001. Revisions made to the referenced plans and procedures should be reviewed to ensure continuing compliance with the restart criterion.

Personnel Protection Procedures

The staff evaluated EPIP 6Q7-102, "Emergency Response Teams," and 6Q7-201, "Emergency Exposure Guidelines," which adequately address specific exposure limits and requirements that must be met prior to personnel exposure that exceeds administrative limits. In addition, both of these EIPs require documentation of present exposure and the dose received during an entry.

The staff reviewed EPIP 6Q7-102, the RAEP, and the SRS Emergency Plan and found them to be consistent with regard to emergency exposure limits for life saving and protection of vital equipment as well as the ERO positions responsible for authorizing emergency exposure. The RAEP indicates that the AEC authorizes emergency exposure for immediate life-saving situations and the TSC Emergency Duty Officer or TSC Health Protection management dictate and authorize all other emergency exposure objectives.

The staff found that there are no specific EIPs that address decontamination of personnel, equipment, and facilities. Decontamination is performed in accordance with the facilities HP procedures. The staff determined that this is acceptable.

The EIPs address performing habitability surveys in the CCR and OSC emergency response facilities, but performing habitability surveys in the Security Tactical Operations Center or other security personnel locations is not specifically addressed. This is an open item. EPIP 6Q7-101, "Activation and Operation of the OSC," indicates that the Protective Force Supervisor is to be kept appraised of radiological conditions in the area and that he/she would inform the HP Coordinator if Security personnel could not be removed from impacted areas. HP could then dispatch personnel to the location(s) to perform monitoring and exposure control briefings.

EPIP 6Q7-303, "Emergency Equipment and Supplies," identifies the emergency radiation monitoring equipment in the OSC and CCR emergency equipment lockers, which is used to monitor habitability of the facilities. The staff noted that a decontamination room and a decontamination station are maintained within each reactor area and EPIP 6Q7-303, "Emergency Equipment and Supplies," identifies the equipment and supplies necessary for monitoring contamination and decontaminating personnel, equipment, and facilities. The staff also noted that EPIP 6Q7-303 identifies what appear to be adequate numbers and types of respiratory protection equipment in the emergency equipment lockers. The Fire Protection Manual referenced in the SRS Emergency Plan identifies the resources of the SRS Fire Department, which has the mechanism for recharging breathing air bottles during an emergency (Reference 26).

The staff reviewed the RAEP and noted that an adequate system of accountability appears to be addressed, including an accountability procedure, three types of rally points, procedures that address notification of personnel, discussion on the protective action requirements for essential and non-essential personnel, and a procedure with provisions for conducting evacuation drills.

The staff reviewed EPIP 6Q7-202, "Accountability and Evacuations," and noted that while protected area accountability using the key-card reader system appears adequate to obtain 30-minute accountability, the procedure does not address protected area personnel accountability in the event that the crash gates are used or the key-card reader fails. This is an open item. Non-essential personnel, including visitor and temporary personnel, are normally evacuated from the Protected Area at the first declaration of an emergency condition.

The staff examined EPIP 6Q7-202, "Accountability/Evacuation," and determined that it addresses the protective action requirements of sheltering and evacuation for reactor personnel. The staff attended GET training and noted that the guidance provided in GET training adequately addresses essential and nonessential personnel protective actions. The staff also noted that EPIP 6Q7-002 addresses the AEC's considerations for determining if evacuation is required and allows for AEC discretion based on existing conditions. Additionally, this procedure indicates that the resources and personnel required to support coordination of the evacuation can be obtained through the TSC.

The staff evaluated EPIP 6Q7-202 and noted that it provides clear guidance on personnel actions for sheltering or evacuation of the facility's protected area. The staff walked down K-Area evacuation routes and noted that they were clearly marked and free of obstacles. The staff could not find a procedure or document that identified permanent evacuation sign posting locations or provisions for periodic inspections to ensure the signs and routes were maintained in a readiness condition. This is an open item.

The staff noted that fire fighting support and response has been adequately prearranged. In a DOE appraisal, "DuPont Operations Emergency Preparedness Program Comprehensive Appraisal Report" (Reference 27), the staff noted that the SRS Fire Department has 16 people and several fire trucks on duty every shift and is capable of responding to any site facility. The staff also noted that the SRS Fire Protection Program requires that each facility have a Fire Protection Plan and that fire response personnel attend periodic training and participate in response drills. There is also an agreement in effect between SRS and the City of Aiken to supply fire suppression assistance to each other in the event of a serious fire, and WSRC has invited the Aiken Fire Department to onsite briefings (References 28 and 29). Training of onsite fire fighters is a continuing activity. Within the last year, two briefings were provided to representatives of Aiken, Augusta, and Richmond County fire departments at SRS. These briefings addressed the support functions that offsite personnel and equipment would perform outside of radiation controlled areas. The staff has further established that if equipment were needed inside radiation controlled areas, it would be manned by SRS personnel.

The staff noted that all personnel entering the Limited Area Boundary are required to wear dosimetry. This indicates that essential security personnel have devices that continuously monitor exposure. The staff reviewed the RAEP and EIPs 6Q7-002, "CCR Staff Actions for Declared Emergencies"; 6Q7-101, "Activation and Operation of the OSC"; 6Q7-202, "Accountability/Evacuation"; and EPIP 6Q7-303, "Emergency Equipment and Supplies," and determined that the intent of the restart criterion is met on the following basis.

- o Even though monitoring of security personnel is not specifically addressed, the Protective Force Supervisor maintains constant communications with the OSC and CCR.
- o The OSC HP Coordinator has the responsibility for keeping the Protective Force Coordinator and AEC apprised of radiological concerns and recommendations.

- o As changing emergency conditions information is obtained, the AEC directs dispatch of OSC response teams. For example, if security personnel must remain in an area of unknown hazards, a team of HP Inspectors would be dispatched to survey the areas and provide protective measure guidance.

Closure of this restart criterion is dependent on correction of the items listed above and on formal approval of the SRS Emergency Plan. Revisions made to the referenced procedure should be reviewed to ensure continuing compliance with the restart criterion.

Emergency Drills and Exercises

The staff determined that the SRS Emergency Plan, Section L, "Drills and Exercises"; RAEP Section L, "Drills and Exercises"; EPIP 6Q-006, "Drills and Exercises" (Reference 30); and EPIP 6Q7-304, "Drills and Exercises," establish a site and reactor area EP drill and exercise program. The reactor area is required to conduct an annual exercise to test and verify the adequacy of the emergency plans, implementing procedures, emergency response facilities, equipment, communications systems, and the capabilities of the ERO. The SRS Emergency Plan, the RAEP and associated EPIPs for emergency drills, and exercise program requirements for the reactor area have been reviewed and found to adequately address this restart criterion.

Closure of this restart criterion are dependent upon formal approval of SRS Emergency Plan. Revisions made to referenced plans and procedures should be reviewed to ensure continuing compliance with the restart criterion.

Sitewide Emergency Program

The staff reviewed the SRS Site Emergency Plan, the RAEP, the 6Q response EPIP Manuals and conducted telephone interviews and determined that the RAEP is an integral part of the SRS Sitewide emergency program, patterned after and coordinated with the SRS Emergency Plan. The SRS Emergency Plan dictates that both plans require DOE approval. The SRS Emergency Plan has met the various parts of this restart criterion, except as noted.

The staff determined that there are provisions in the SRS Emergency Plan training section for providing personnel in offsite protective action decision-making ERO positions with training on the relationship between reactor conditions and protective measures.

A comparative review of the list of ERO positions and responsibilities and the training matrix of ERO positions versus EP training assignments indicates that all ERO positions involved with offsite protective action decision-making are required to attend training on emergency classification, protective action recommendations, dose assessment, and accident assessment.

The staff reviewed the SRS Emergency Plan, the RAEP, and the associated EPIPs that address coordination and communication between the TSC/EOF and the Reactor Area ERO and emergency facilities. The staff determined that communications systems outlined in the plans and procedures appear adequate

for emergency communications. Additionally, the ERO positions and responsibilities for coordination of the response effort appear to meet the restart criterion.

Provisions to assure the availability of the engineering documentation and the emergency preparedness documentation to the TSC and EOF for coordination with reactor area personnel in the mitigation of accidents have not been addressed. This is an open item.

The staff noted that there is a need for clarification on specific communications links for technical information transfer between the TSC and the Reactor Area and the EOF and HQ Emergency Operations Center. This is an open item.

The staff determined that there are no formal procedures for coordinating the transfer of Reactor Area evacuees from the assembly area or rally points to a safe area. The existing EIPs indicate that coordination for the transfer would be performed by the Site Support Director by contacting Works Engineering. There are several non-emergency designated vehicles available through Works Engineering, but neither Works Engineering nor the TSC emergency notification procedure has a formal emergency call-out list of vehicle drivers. This is an open item.

The staff determined that the SRS evacuation host locations (e.g., P-Reactor Disassembly Area) are shown in the SRS Emergency Plan. A site evacuation procedure or equivalent is needed, including the establishment of the mechanism for transporting evacuees from rally points to a host location, identification of evacuation routes from each SRS facility, and accommodations at the host facilities. This is an open item.

The staff determined that the agreements and provisions for providing transportation and medical treatment to contaminated and injured personnel is adequate. The staff noted that there is an agreement between SRS and the Eisenhower Army Medical Center (EAMC) at Fort Gordon that ensures that radiologically contaminated patients who need more assistance than the SRS Medical Department can provide will be accepted for treatment.

In April 1990, the staff ascertained that the SRS Medical Department maintains a staff of 8 doctors, 19 registered nurses, and 16 emergency medical technicians to provide continuous medical coverage. The staff also noted that three ambulances are available onsite for the transportation and emergency medical treatment of injured personnel.

The staff reviewed the SRS Emergency Plan, the RAEP, and their related implementing procedures concerning non-delegable responsibilities for emergency classification, protective action requirements, and offsite notification and found they were consistent with the line of succession of the emergency coordinator position. The staff determined that the subjects are addressed adequately to meet the restart criterion.

The staff noted that the SRS emergency notification list (i.e., DPSOP 287, Appendix G) and its draft replacement, 6Q12 Manual (Reference 31), which is located in the TSC Communications Center, identify affected program personnel who are called out for reactor area emergencies. This list indicates that the identified personnel have reactor safety analysis expertise and report to the TSC, which adequately addresses this restart criterion.

The SRS Emergency Plan section on emergency preparedness training has been reviewed by the staff, and specific dose assessment training is listed on the training matrix for all ERO positions involved with dose assessment. This training includes operations of the WINDS and dose assessment programs.

The staff reviewed the SRS Emergency Plan and its associated implementing procedures and noted that the responsibilities and authority between DOE/SR, DOE/SRSPO and the DOE Contractors, as specified in the EOF, have been defined and documented. Although DOE/SRSPO responsibility is not separately addressed, SRSPO management staff would function as the DOE Operations Advisor and HQ Communicator in the event of an emergency at one of the reactor facilities. The staff reached this conclusion on the basis of its review of the DOE-SR Emergency Response Organization Personnel Assignments list, which indicates that the SRSPO Director would be the primary EOF Operations Advisor during a reactor incident and the SRSPO Technical staff would be assigned the EOF DOE-HQ Communicator's position.

The staff determined that WSRC's Emergency Preparedness Section has developed an administrative procedure that addresses the assignment, distribution, and testing of pager and page-comm type radio receivers for ERO personnel. The staff determined that there is no corresponding procedure that addresses assignment and distribution of pagers or page-comm type radio receivers to remote work groups. This is an open item. There are two action tracking systems that address this deficiency: (1) DOE/OEA Finding and Commitment Tracking System (FACTS) Item 331 and (2) WSRC Commitment Management System (CMS) Item 2452. A policy review board was scheduled to address this issue and assign program responsibility in early February 1991. The staff will review the resolution of this issue and report on its evaluation in a supplement to this report.

The staff determined that WSRC has developed a procedure for periodically testing the pager and page-comm radio receiver systems via the TSC Communication Center; however, the WSRC 12Q Procedures Manual, which contains the procedure, has not been approved. This is an open item. The policy for assignment and distribution of pagers or page-comm type radio receivers to remote workers must be determined, implementation procedures must be developed, and training must be conducted. This is a personnel safety issue and should be addressed as soon as possible. The procedure for periodic testing of the pager and page-comm systems needs to be approved and implemented.

Closure of this restart criterion is dependent on correction of the items listed above and on formal approval of the SRS Emergency Plan. Revisions made to the referenced plans and procedures should be reviewed to ensure continuing compliance with the restart criterion.

15.4 Conclusion

Based on the above evaluation the DOE staff concludes that Emergency Preparedness is an open item. Formal DOE approval of the SRS Emergency Plan and EPIP 6Q7-001 will considerably reduce the number of deficiencies noted in this report and the staff's review of any revisions made to referenced documentation will ensure continuing compliance with restart criteria. If the additional deficiencies are adequately addressed, the Emergency Preparedness Program will meet the restart criteria.

The staff will evaluate the open items discussed above and document the results of their review in a supplement to the SER.

15.5 References

1. WSRC, "SRS Emergency Plan 6Q," Revision 0, June 1, 1990 (draft).
2. WSRC, Procedure MRP 4.08, "Events, Conditions, and Concerns Determination," January 1, 1991.
3. WSRC, SRS Emergency Plan Implementing Procedure EPIP 200, Revision 0, October 1, 1990 (draft).
4. WSRC, "Reactor Area Emergency Plan 6Q7," Revision 0, August 1, 1990.
5. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-302, "Emergency Response Organization and Emergency Training," Revision 0-A, November 15, 1990.
6. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-101, "Activation and Operation of the OSC," Revision 0, August 1, 1990.
7. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-102, "Emergency Response Teams," Revision 0, August 1, 1990.
8. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-002, "CCR Staff Actions for Declared Emergencies," Revision 0, August 1, 1990.
9. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-201, "Emergency Exposure Guidelines," Revision 0, August 1, 1990.
10. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-202, "Accountability/Evacuation," Revision 0, August 1, 1990.
11. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-203, "Personnel Injury," Revision 0, August 1, 1990.
12. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-205, "Potassium Iodide (KI) Administration," Revision 0, August 1, 1990.
13. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-303, "Emergency Equipment and Supplies," Revision 0-A, November 15, 1990.

14. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-002, "CCR Staff Actions for Declared Emergencies," Revision 0, August 1, 1990.
15. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-001, "Emergency Classification," Revision 0, August 1, 1990.
16. WSRC, Reactor Area Emergency Plan Implementing Procedure 6Q7-304, "Drills and Exercises," Revision 0-A, November 15, 1990.
17. WSRC, Reactor Area Abnormal Conditions Control Procedures (100-700) Series.
18. WSRC, "K-Reactor Area Safety Analysis Report," July 17, 1989.
19. WSRC, Reactor Area Emergency Preparedness Training Matrix, June 1990.
20. WSRC, DPSOL 292-419 "Plant HP Emergency Equipment Audit."
21. WSRC, Medical Department Manual 3Q-7, "First Aid and Nursing Procedures," 4008, "Checking Treatment Room and Stocking Emergency Cabinet."
22. WSRC, Standing Instruction No. II-8, April 1990.
23. WSRC, DPSOL-100-4000-PLK, "Emergency Action Levels," Revision 4, February 28, 1990.
24. WSRC, WSRC-TR-90-42-126, "Technical Basis for EPIP 6Q7-001, Emergency Classification (U)," November 11, 1990.
25. WSRC, WSRC-TM-90-14, "WINDS User's Manual," December 10, 1990.
26. WSRC, DPSOL 105-1912-PLK, "Inspection of Emergency Breathing Apparatus," Revision 11, November 7, 1988.
27. U.S. Department of Energy Office of External Affairs, "DuPont Operations Emergency Preparedness Program Comprehensive Appraisal Report," Appendix B, Item III-1, September, 1988.
28. Contract Number AT (381)- 220, dated 10 May 1960, between the United States Atomic Energy Commission and the City of Aiken, SC.
29. Letter, Paul J. Hagelston, Director Safety And Technical Services Division, Savannah River Operations Office, United States Atomic Energy Commission to Mr. H. Odell Weeks, Aiken, S.C., dated November 22, 1968.
30. WSRC, SRS Emergency Plan Implementing Procedure EPIP 6Q-006, Revision 0, "Drills and Exercises," June 1, 1990 (draft).
31. WSRC, DPSOL 287, Appendix G, "Westinghouse Savannah River Site Emergency Management Notification List for Operations/ Security Alert/Emergency Including Plant Vogtle Power Reactor," Revision 125, November 17, 1989.

16 OPERATIONAL READINESS REVIEW (ORR)

16.1 Background

Management requires an accurate, auditable assessment of the readiness-to-operate status of the facility prior to permitting reactor startup. The contractor operational readiness review (ORR) provides an assessment that is broadly based and includes consideration of safety, environment, technical, operational, programmatic, and administrative activities. The ORR is independent of other elements of line management's activities to the extent necessary to provide an unbiased, technically competent assessment. The assessment is based on a thorough and auditable review of appropriate hardware, documentation, personnel, management controls, and programmatic elements of the facility. The ORR is intended to be an independent source of information that can be used by line management when making startup decisions.

Westinghouse Savannah River Company (WSRC) committed to issuing and implementing a readiness evaluation plan and documenting the results. WSRC management also recognized the need for both a readiness evaluation by responsible line personnel and a second, independent evaluation.

16.2 Restart Criteria

(1) General

- (a) WSRC shall perform an Operational Readiness Review (ORR) to provide an auditable assessment of reactor readiness for operation.
- (b) The ORR shall be based on thorough reviews of appropriate documentation, hardware systems, personnel training, management, and administrative controls. The ORR shall provide a structured methodology for determining that each reactor is ready for safe startup and operation.
- (c) The ORR shall not replace contractor line management's primary responsibility for project completion, quality assurance, technical adequacy, or resolution of safety or process-related considerations, etc. Rather, an ORR shall provide a review function to ensure that all necessary actions by the responsible line organization have been completed.
- (d) WSRC shall establish an ORR program to govern all ORR activities. The ORR program shall contain the following elements:
 - o ORR organization with adequate representation from the line organization responsible for reactor restart
 - o ORR plan and procedures
 - o independent evaluation of the ORR

(2) Requirements

- (a) WSRC shall establish procedures to define: (1) the ORR organization, (2) duties and responsibilities of ORR management and team/reviewer participants, and (3) qualifications of affected members of the ORR organization. WSRC shall also identify staffing to meet these requirements.
- (b) WSRC shall meet the following criteria:
 - o Provide an ORR plan and procedures.
 - o Provide the overall scope of the ORR associated with reactor operation.
 - o Develop an ORR tree based on industry recognized analytic-tree methodology to ascertain adequacy and depth of coverage of the ORR scope.
 - o Develop checklists and acceptance criteria from items B and C.
 - o Develop training requirements for ORR participants who perform operational readiness reviews.
 - o Develop a tracking system to ensure that all ORR activities are completed.
 - o Describe the method to be used for resolving ORR findings either in the ORR plan or in applicable procedures.
 - o Provide closure packages for each area reviewed.
 - o Provide a final ORR report and the schedule for completion and delivery of the final report to DOE.

(3) Independent Evaluation

WSRC shall perform an independent evaluation/assessment of the readiness for restart. The independent evaluation group shall demonstrate its independence from the ORR organization and develop an evaluation plan to be used to conduct the independent evaluation of the reactor restart readiness. The independent evaluation group shall also assess the ORR effort including (as a minimum) evaluation of the ORR plan, ORR procedures, the ORR tree and checklist items, the ORR checklist item acceptance criteria, ORR team members qualification and training, and ORR closure packages.

(4) DOE-WSRC Interface

- (a) DOE management approval of the ORR plan and the independent evaluation plan shall be required. WSRC shall communicate with DOE on progress of the ORR via periodic meetings and reports.

- (b) WSRC shall develop an ORR schedule with identifiable milestones. DOE concurrence shall be obtained prior to proceeding past identified milestones.
- (c) WSRC shall notify DOE of any significant findings discovered during the ORR. As a minimum, reported findings shall include those identified in the notification and reporting criteria.
- (d) Upon completion of each system readiness package, WSRC shall notify DOE.

16.3 Evaluation

The staff reviewed OPS-SAM-890008, "Reactor Restart Operational Readiness Review Plan" (Reference 1); OPS-SAM-890009, "Reactor Restart Operational Readiness Review Procedure" (Reference 2); and supporting documents to determine the effectiveness of the ORR process to determine the reactor readiness to operate. DOE approved the plan and procedure in June 1990. The staff also reviewed ESH-RSE-90072, "Reactor Operational Readiness Evaluation Plan" (Reference 3), and WSRC-Q4, RSES 203, "K-Reactor Operational Readiness Evaluation Procedure" (Reference 4), which performs an independent evaluation of the ORR process. The Reactor Safety Evaluation Section's (RSES) Operational Readiness Evaluation (ORE) Team of the Environmental Safety, Health, and Quality Assurance (ESH&QA) Division performs this independent evaluation of the ORR process. DOE approved the ORE plan and procedure in June 1990.

The staff found that the Readiness Review Board (RRB) had been established and were meeting on a regular basis. The staff attended most of the meetings and found that, at first, the RRB was not providing the necessary guidance and control. In time, the RRB became better organized and took control of the ORR process such that a thorough ORR is being conducted.

An additional assurance of readiness was provided through the RSES ORE Team. Members of the ORE were interviewed and observed attending the RRB meetings and evaluating the ORR process.

OPS-SAM-890008, Section IV, states that the ORR organization consists of a Readiness Review Board, a support staff for the Board, and checklist item reviewers. The RRB is composed of a chairman, a vice chairman, a staff manager and five other members with alternates. The Deputy General Manager, Reactors, is the chairman. The RRB will also have several non-voting observers from other organizations such as RSES and QA. Exhibit 1 of OPS-SAM-890008 summarizes the ORR organization. In addition, OPS-SAM-890009, Sections 5.1 and 6.1, further describe the organization of the RRB and RRB staff. The RRB staff will nominally be comprised of a manager, several technical personnel, a database expert, and clerical support.

OPS-SAM-890008, Section V, describes the responsibilities of the following personnel and groups: (1) Deputy General Manager, Reactors, (2) RRB, (3) RRB staff, (4) checklist item reviewers, (5) observers, (6) Manager, Strategic Programs, and (7) independent oversight agency. In addition, OPS-SAM-890009

(Sections 5.2, 6.2, 7.0 and 8.0) further describes the responsibilities of the RRB, RRB staff, checklist item reviewer, and other groups.

OPS-SAM-890008, Section IV, and OPS-SAM-890009, Section 6.3, state the qualifications of the ORR team members shall be a technical degree with 3 years nuclear experience or 10 years nuclear experience and no degree.

OPS-SAM-890008, Section VI.C, and OPS-SAM-890009, Section 6.4, state that the RRB staff manager is responsible for training development, and execution. The training is to center around ORRs, management oversight and risk tree, and investigative team-type efforts. Each member of the ORR team will receive formal training on the contractor's ORR Plan and Procedure, the DOE-SR Order and Manual on ORRs, the Reactor Operations Management Plan, and the DOE Safety Evaluation Report. In addition, at least one member of the team will have formal training in MORT.

The overall scope of the ORR is described in OPS-SAM-890008, Section II. Each reactor ORR is to be performed and documented to provide added assurance that additions or changes to the facilities, processes, or administrative controls: (1) are implemented as designed, (2) can be operated safely and securely, (3) will perform as designed, (4) will be operated or used by trained and qualified personnel, and (5) have received adequate consideration so that operation will not create undue risk to the employees, the public, or the environment.

OPS-SAM-890009, Section 1.3, states that the ORR process has been developed using several documents, including the generic Management Oversight and Risk Tree (Whitsett ORR Handbook); Institute of Nuclear Power Operations, INPO 85-001 "Performance Objectives and Criteria for Operating and Near-term Operating License Plants"; and SR Order 548X.1/548X.1A (draft) "Operational Readiness Reviews."

OPS-SAM-890008, Section VI.C, states that the RRB staff will develop the checklist of the reactor ORR using the generic Management Oversight and Risk Tree. Existing hardware, administrative controls, and personnel items associated with reactor operation are to be listed and related to the appropriate section of the tree to create the checklist. OPS-SAM-890009, Section 11.0, provides the instructions for preparing and using the checklist. Exhibit 3 identifies the areas of reviews that will require checklists.

OPS-SAM-890008, Section VI.D, states that specific ORR acceptance criteria are to be prepared by a checklist item reviewer with a peer review of the acceptance criteria by the RRB staff. The RRB will then review and approve each acceptance criterion. OPS-SAM-890009 Section 12.0 provides the instructions for developing and using the acceptance criteria. Generic acceptance criteria are to be used for many of the hardware checklist items to maximize the effectiveness of the ORR effort. A list of specific basic elements for each hardware checklist item will be developed for review against that related generic restart criteria.

OPS-SAM-890009, Section 13.0, provides instructions for the preparation of the Reactor Readiness Package (RRP). This package is prepared by the checklist

item reviewer and will contain documentation that the item reviewed meets the acceptance criteria. The minimum documentation required to be included in the RRP is given in Section 13.1.4. The RRP is to be reviewed by the RRB staff for thoroughness and completeness and approved by the RRB. Section 6.2.14 requires the RRB staff to transmit the copies of the open items, open item disposition, and punchlists to the DOE staff. The RRB staff is also required to transmit copies of the acceptance criteria and readiness packages to ORE and DOE as requested. Section 17.5 requires the RRB staff to notify the DOE staff upon completion of each system readiness package and when significant findings are discovered during the ORR.

The instructions for recording and tracking punchlist items are provided in OPS-SAM-890009, Section 14.0. The punchlist will ensure that all ORR activities are performed. A punchlist has been established for tracking activities that are to be performed prior to restart and another for ORR activities that can be performed after restart. It is the responsibility of the RRB staff to prepare and maintain the database for the punchlist items. Appropriate line management is to notify the RRB of the completion of a punchlist activity.

Any aspect of a checklist item found during the course of the review that does not meet all or part of the approved acceptance criteria is defined as an open item. OPS-SAM-890009, Section 15.0, provides instructions for the disposition of open items.

OPS-SAM-890009, Section 17.0, addresses the reporting requirements for the ORR. Reports on the ORR status, problem areas, and success areas are to be part of the RRB meeting agenda. Periodic written status reports are also to be issued to keep reactor restart management informed about the ORR. The ORR plan requires that DOE receive these periodic ORR reports. In addition, the ORR plan requires DOE approval be obtained for continuance past identified plan milestones. DOE staff periodically attend the RRB meetings to monitor progress and are provided written status reports developed by ORR members.

A detailed ORR report will be prepared and issued for Phase I (fuel load), Phase II (power operation), and follow-up phases. Each report is required by OPS-SAM-890009, Section 17.4, to contain a summary, introduction, description of the ORR process, results, approvals, and appendices (as needed). The schedule for each reactor ORR is interrelated with and part of the overall reactor restart schedule. The ROMP provides detailed schedule information for each reactor ORR. The DOE is on distribution for receipt of these ORR reports.

An independent evaluation of the ORR process is to be performed by the Quality Assurance Department and by the Reactor Safety Evaluation Section (RSES) Operational Readiness Evaluation (ORE) team to provide additional assurance of readiness, as stated in Sections II and V.G of OPS-SAM-890008. The ORE team is not to replace the line organization's responsibility in the review and/or disposition of items. The independence of this evaluation is also described in the ORE plan, ESH-RSE-90072 "Reactor Operational Readiness Evaluation Program Plan," and the ORE procedure WSRC-Q4, RSES 203 "K-Reactor Operational Readiness Evaluation Procedure." Sections 1, 2, and 3 of the ORE plan require

the ORR process and implementation described by the ORR plan and procedure to be reviewed for adequacy. Section 4 of the ORE plan for reviewing the ORR discusses the evaluation of the ORR tree and checklist items. The ORR checklist acceptance criteria is reviewed for adequacy in accordance with Sections 5 and 6 of the ORE plan. The qualifications and training of ORR members are to be reviewed in accordance with Sections 7, 8, and 9. The ORE evaluation of ORR closure packages is discussed in Sections 14 and 15.

16.4 Conclusion

The ORR program review is now complete. The DOE staff has determined that the ORR program satisfactorily addresses the restart criteria specified in the SER.

16.5 References

1. WSRC, OPS-SAM-890008, "Reactor Restart Operational Readiness Review Plan," Revision 4.
2. WSRC, OPS-SAM-890009, "Reactor Restart Operational Readiness Review Procedure," Revision 4.
3. ESH-RSE-90072, "Reactor Operational Readiness Evaluation Plan."
4. WSRC, WSRC-Q4, RSES 203, "K-Reactor Operational Readiness Evaluation Procedure."