UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-424/89-25 and 50-425/89-29

Licensee: Georgia Power Company
P.O. Box 1295
Birmingham, AL 35201

Docket Nos.: 50-424 and 50-425 License Nos.: NPF-68 and NPF-81

Facility Name: Vogtle 1 and 2

Inspection Conducted: August 19 - September 15, 1989

Inspectors: _____ 10-11-89
John F. Rogge, Senior Resident Inspector Date Signed

_____ 10-11-89
Ronald F. Aiello, Resident Inspector Date Signed

Accompanied by: Robert D. Starkey, Randy Musser (September 7/8, 1989)

Approved By: _____ 10/12/89
Kenneth E. Brockman, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope:

This routine inspection entailed resident inspection in the following areas: plant operations, radiological controls, maintenance, surveillance, security, technical support, and quality programs and administrative controls affecting quality.

Results:

One cited violation and one non-cited violation were identified. The cited violation was in the area of technical support for failure to adequately certify, pursuant to 10 CFR 55.23, that a medical examination had been performed which met 10 CFR 55.21 requirements (paragraph 2.b(9)). The non-cited violation was in the area of maintenance for failure to maintain 10 CFR 50.49 equipment qualification on three temperature RTDs due to missing O-rings (paragraph 3.b(2)(c)).

The inspection report discusses one strength in the area of Radiation Controls regarding the process of performing investigations (paragraph 2.b(4)).

Weaknesses were identified in the technical support area regarding the medical qualification process of licensed operators (paragraph 2.b(9)) and in the area of plant operations, regarding the updating, completeness, and use of qualification lists by the Shift Supervisors (paragraph 2.a).

DETAILS

1. Persons Contacted

Licensee Employees

*J. Aufdenkampe, Plant Engin[illegible]ing Supervisor
*G. Bockhold, Jr., General [illegible]er Nuclear Plant
C. Coursey, Maintenance Su[illegible]ntendent
*G. Frederick, Safety Audit and Engineering Group Supervisor
H. Handfinger, Manager Maintenance
K. Holmes, Plant Training and Emergency Preparedness Manager
*W. Kitchens, Assistant General Manager Plant Operations
*R. Legrand, Manager Chemistry and Health Physics
*G. McCarley, Independent Safety Engineering Group Supervisor
*A. Mosbaugh, Plant Support Manager
W. Mundy, Quality Assurance Audit Supervisor
*R. Odom, Nuclear Safety and Compliance Manager
*J. Swartzwelder, Manager Operations

Other licensee employees contacted included technicians, supervisors, engineers, operators, maintenance personnel, quality control inspectors, and office personnel.

*Attended Exit Interview

An alphabetical list of acronyms and initialisms is located in the last paragraph of the inspection report.

2. Operational Safety Verification - (71707)(93702)

Both units operated near 100% power during this inspection period except for minor power reductions for maintenance. On September 13, with Unit 2 reduced to 90% power for replacement of a feedwater pump power supply, the loss of a second feedwater pump occurred. Unit 2 power was reduced to within the capacity of a single feedwater pump by the operators. The inspector subsequently reviewed the transient on the simulator.

a. Control Room Activities

Control Room tours and observations were performed to verify that facility operations were being safely conducted and within regulatory requirements. The following attributes, as appropriate at the time of the inspection, were included:

- Proper Control Room staffing;

- Control Room access and operator behavior;

- Adherence to approved procedures for activities in progress;

- Adherence to technical specification limiting conditions for operations;

- Observance of instruments and recorder traces of safety related and important-to-safety systems for abnormalities;

- Review of annunciators alarmed and action in progress to correct same;

- Control Board walkdowns;

- Safety parameter display and the plant safety monitoring system operability status;

- Discussions and interviews with the On-Shift Operations Supervisor, Shift Supervisor, Reactor Operators, and the Shift Technical Advisor (when stationed) to determine the plant status and plans, and to assess operator knowledge;

- Review of the operator logs, unit logs, and shift turnover sheets

During two separate deep backshifts, September 3 and 4, 1989, the inspector requested the Shift Supervisors on each unit to demonstrate that all watchstanders were currently qualified. On each occasion, the Shift Supervisors utilized qualification lists located in the control room. These lists were dated July 10 and 20, 1989, depending on the watch position. The inspector questioned the supervisors on their confidence in using sheets that were dated almost two months earlier. After receiving assurance, the inspector pointed out that the off-going reactor operator on September 3, 1989, had been struck off the list as a qualified operator. This operator had been removed from licensed duties on July 31 and was reinstated on August 14, 1989. Further discussions with the Shift Supervisors revealed that they did not normally verify their watchstanders because they felt they were aware of the qualification status. The inspector identified this as a weakness to operations management and, in addition, noted that individual license conditions were not included on the lists. Further followup on the ability of shift personnel to verify qualifications will be conducted during the next inspection.

No violations or deviations were identified.

b. Facility Activities

Facility tours and observations were performed to assess the effectiveness of the administrative controls established by the licensee. Direct observation of plant activities, interviews and discussions with licensee personnel, independent verification of

safety systems status and LCOs, observation of licensee meetings, and review of facility records were accomplished by the inspectors. During these inspections, the following objectives were achieved:

(1) Safety System Status (71710) - Confirmation of system operability was obtained by verification that flowpath valve alignment, control and power supply alignments, component conditions, and support systems for the accessible portions of the ESF trains were proper. The inaccessible portions were confirmed as availability permitted. An additional indepth inspection of the AFW system, with emphasis on Unit 2, was performed to compare the system lineup procedure with the plant drawings, the as-built configurations, and remote and local valve indications. One system walkdown was expanded to include hangers and supports. A weakness was identified in the labeling system. The labels on the components, especially electrical, were not consistent with the nomenclature written in the procedure. The inspector verified that the lineup was in accordance with license requirements for system operability. The nomenclature problem was brought to the attention of the operations management. The licensee is reviewing the inspectors comments.

(2) Plant Housekeeping Conditions - Storage of material and components and cleanliness conditions of various areas throughout the facility were observed to determine whether safety and/or fire hazards existed.

(3) Fire Protection - Fire protection activities, staffing, and equipment were observed to verify that fire brigade staffing was appropriate and that fire alarms, extinguishing equipment, actuating controls, fire fighting equipment, emergency equipment, and fire barriers were operable.

On September 14, 1989, the inspector observed a fire drill conducted in the diesel generator building. The drill critique was also attended.

(4) Radiation Protection - Radiation protection activities, staffing, and equipment were observed to verify proper program implementation. The inspection included review of the plant program effectiveness. Radiation work permits and personnel compliance were reviewed during the daily plant tours. Radiation Control Areas were observed to verify proper identification and implementation.

On September 6, 1989, the inspectors observed the health physics processing of a visitor, following the detection of Cs-137 during an exit whole body count. The visitor's entrance whole body count had not detected the presence of Cs-137. A second

exit whole body count, with clothes removed, verified the initial detection of Cs-137. The presence of CS-137 was calculated to be 5-7 nCi. (The minimum detectable amount for this isotope of 9 nCi). The health physics review determined that the individual had been contaminated by Chernobyl fallout and had alarmed monitors at Yugoslavian nuclear facility power plants. The inspectors direct observation of the investigation concluded that the health physics department utilized professional and competant practices in performing investigations.

(5) Security - Security controls were observed to verify that security barriers were intact, guard forces were on duty, and access to the Protected Area was controlled in accordance with the facility security plan. Personnel were observed to verify proper display of badges and that personnel requiring escort were properly escorted. Personnel within Vital Areas were observed to ensure they had proper authorization for the area.

Equipment operability or proper compensatory activities were verified on a periodic basis. On two occasions, the inspector observed actual response to these vital area door alarms. In each case, the inspector determined that the guard force responded appropriately.

(6) Surveillance (61726) - Surveillance tests were observed to verify that approved procedures were being used, qualified personnel were conducting the tests, tests were adequate to verify equipment operability, calibrated equipment was utilized, and TS requirements were followed. The inspectors observed portions of the following surveillances and/or reviewed completed data against acceptance criteria:

Surveillance No.	Title
14225-2, Rev. 3	Operations Weekly Surveillance Logs
14495-2, Rev. 0	TDAFW Pump Flowpath Verification
14546-2, Rev. 2	TDAFW Pump Operability Test
14802-2, Rev. 1	NSCW Pumps And Discharge Check Valves Inservice Test
14825-1, Rev. 11	Quarterly Steam Generator Blowdown
14825-2, Rev. 2	Quarterly RHR Valve Inservice Test

Surveillance No. (cont'd)	Title
24714-1, Rev. 10	NIS Power Range Channel N-44 Channel Calibration
24813-1, Rev. 10	Loop 4 Delta/Tave ACOT And Channel Calibration

(7) Maintenance Activities (62703) - The inspector observed maintenance activities to verify that correct equipment clearances were in effect, work requests and fire prevention work permits, as required, were issued and being followed, quality control personnel were available for inspection activities, retesting and return of systems to service was prompt and correct, and TS requirements were being followed. The Maintenance Work Order backlog was reviewed, and maintenance was observed and/or work packages were reviewed for the following maintenance activities:

MWO No.	Work Description
18900995	Implement MDD #89-VIM020 (Change TDAFW MOV DC Breakers 1CDIM01, 02, 03, 04, And 05 From 15A To 30A Thermal Magnetic)
18903023	Investigate And Rework Loop 961 Accumulator #1 Pressure Indication
18903753	Repair CVCS Gauge Line Connection
28902244	Investigate And Rework Wiring On The RCP #1 Thermal Barrier Isolation Valve (2HV-19051)

(8) Emergency Preparedness - The inspector conducted a tour of the Emergency Response Center, Onsite Support Center, and Technical Support Center to verify readiness and familiarize the Hatch Resident Inspector with the facilities.

(9) Operator Medical Examinations - The inspector reviewed the facility documentation pertaining to NRC licensed operators in order to evaluate the effectiveness of the licensee's program. All Senior Reactor Operator and Reactor Operator medical documentation (License, NRC Form 396, Licensed Operator Certificate of Medical History and Medical Examination, and associated medical test results) was reviewed. The following items were identified during the inspectors review and were presented to the licensee on September 1, 1989:

(a) 10 CFR 55.27 requires the facility licensee to document and maintain the results of medical qualifications data, test results, and each operator's or senior operator's medical history for the current license period and provide the documentation to the Commission upon request. The inspectors noted, during their review, that information was randomly filed within the training record of each individual. In addition, deficiencies were noted within individual records, such as:

- Facility Operator Reports - ANSI Standard 3.4 - 1983, Section 3.2 requires the facility to forward to the medical examiner a report on each employee referred for a nuclear reactor operator medical examination, prior to the examination. This report is required to include information specified by the designated medical examiner and should address areas such as work performance, attendance, and behavioral changes noted since the previous review. The inspectors noted that this document was not included in all records.

- Medical Examination - ANSI Standard 3.4 - 1983, Section 5.4.5, Eyes, requires near and distant visual acuity of 20/40 in the better eye, corrected or uncorrected. Two different operator medical examinations indicated that a better eye existed within the 20/40 uncorrected range; however, the medical examiner recommended that the respective licenses be conditioned to require corrective lenses while performing licensed duties. This recommendation was certified by the facility on NRC Form 396 to the NRC and the licenses were subsequently issued. The inspectors considered this to be an incorrect interpretation of the standard by the medical examiner combined with an inadequate facility review of the examination results. Since neither operator should be restricted, this item was identified to the licensee for review.

- Medical Examination - The form utilized by the facility consists of two basic parts and is modeled after an older version of NRC Form 396. Part 1 consists of Blocks 1 through 38 for the individual to fill in and sign. Part 2 consists of Block 39, Physical Examination Blocks 1 though 25, and a medical examiner's certification of the persons health along with recommended license conditions. The inspectors noted that no elaboration past describing the condition was provided in Block 37, which asks the

individual to describe for each "yes" answer (Blocks 5-33), the condition and explain why this matter would not affect their ability to function as a facility operator. In one instance, the medical examiner made no entries in Block 39. Block 39 asked the physician to summarize and elaborate on the medical history provided by the individual in Blocks 1 through 38. With no entry, the inspector could not conclude if the examiner had reviewed the history in coming to a final determination. Other examples of missing information were identified to the licensee. These oversights are not being picked up by the medical examiner or facility. These errors serve to illustrate that no effective facility review is apparent prior to certification to the NRC on Form 396.

- Other Test Results - The overall content of medical documentation varied with each record. The facility provides medical testing onsite and forwards the results to the medical examiner. Hearing tests, blood tests, and all other documents generated pursuant to the examination should be maintained, as specified in 10 CFR 55.27. However, some records contained only the medical summary sheet. While the inspectors saw no need to have to ask for further information, this was indicative of the facility not having a program for ensuring they meet the requirements to maintain medical records.

(b) 10 CFR 55.23 requires the facility, in part, to certify that a physician has conducted the medical examination as required by 10 CFR 55.21, and when the certification requests a conditional license based on medical evidence, the medical evidence must be submitted on NRC Form 396. The inspectors noted that one medical examination, performed on March 2, 1989, documented an eye exam with a distant visual acuity uncorrected of 20/100 and 20/200, left and right eye respectively. The applicant's vision, however, was correctable to within ANSI 3.4 requirements. Neither the applicants NRC Form 396, dated May 23, 1989, nor the medical exam from the doctor, dated March 2, 1989, indicated a need for a license, condition regarding corrective lenses. This resulted in a license, dated July 11, 1989, without the proper condition. The individual had not answered "yes" to the Part 1-Block 25 question, asking if eyeglasses were normally worn, nor was Part 1-Block 37, asking for details, filled in. The inspector identified to the facility that the individual's license did not have a proper restriction. The facility initiated

a letter to the individual and his shift management requiring the wearing of corrective lenses while performing licensed duties. In addition, they verified that the operator had not performed licensed duties without corrective lenses. The facility is planning to submit an updated NRC Form 396 within 30 days. Since the operator had always performed his duties with corrective lenses the significance of the issue is low; however, it is indicative of a lack of facility oversight in monitoring the medical examination process.

The licensee promptly responded to the inspector's findings by conducting a review. Their review confirmed the findings and, in addition, identified that:

(a) Two sets of medical records were being maintained. The first set was in the training records and the second set was in the personnel records.

(b) The originals were forwarded to training with a copy to personnel.

(c) The plant nurse served as the site coordinator of the documentation and medical examination.

(d) In the case of the operator that had been certified to need no restriction, the plant nurse had received a pre-signed certificate which did not include laboratory data (block 23), summary evaluation (block 24), final medical conclusion (block 25), or license conditions. The original was sent to training and a copy was retained by the nurse. The routine nursing practice was to then obtain the missing information from the medical examiner and with concurrence, make the entries on the copy. The original, in training, would be telephonically updated. During this process, the original was annotated with "No Restrictions," however, the personnel copy showed "Corrective lenses must be worn..." restriction. Training prepared the NRC Form 396 based on the incorrect original certificate.

(e) Changes after the medical examiner had signed and dated other examinations were handled in a similar manner.

(f) The plant had never established any formal procedure specifying responsibilities, training requirement, record retention, etc.

On September 13, 1989, the licensee admitted that a violation of 10 CFR 55.23 had occurred in that an improper medical certification had been made to the NRC because it was not based on an adequate and procedurally approved program. The following corrective actions were determined to be applicable:

(a)	Write a letter to the Operator requiring corrective lenses to be worn while on shift.	Completed 9-1-89
(b)	Verify the Operator was never manning the shift without corrective lenses.	Completed 9-8-89
(c)	Meet with Plant Designated Medical Personnel to discuss the accuracy of the Operator Medical records.	Completed 9-11-89
(d)	Submit revised medical record to NRC.	Scheduled to be complete by 10-5-89
(e)	Define in Plant Procedures what consitutes a NRC License Operator Medical Record and ensure future examinations meet these requirements. Review all records during consolidation.	Scheduled to be complete by 11-15-89
(f)	Meet with Plant Designated Medical Personnel and doctor to review ANSI 3.4-1983/Regulatory Guide 1.134 requirements.	Scheduled to be complete by 10-6-89

The above item represents a violation of NRC requirements.

VIOLATION 50-424/89-25-01 and 50-425/89-29-01, "Failure to Adequately Certify Pursuant to 10 CFR 55.23 that a Medical Examination Had Been Performed Which Met 10 CFR 55.21 Requirements."

One violation was identified in paragraph 2.b(9).

3. Review of Licensee Reports (90712)(90713)(92700)

a. In-Office Review of Periodic and Special Reports

This inspection consisted of reviewing the below listed reports to determine whether the information reported by the licensee was technically adequate and consistent with the inspector's knowledge of the material contained within the report. Selected material within the report was questioned randomly to verify accuracy and to provide

a reasonable assurance that other NRC personnel have an appropriate document for their activities.

Monthly Operating Report - The report dated August 8, 1989, was reviewed. The inspector had no comments.

Special Report 50-424/89-02 (Closed) - "Valid Diesel Generator 1B Failure." On July 19, 1989, Diesel Generator 1B was started for the monthly surveillance testing and loaded to approximately 5000 KW when the operator noticed spikes in the current and KVAR meter readings. The generator was taken off-line and the field voltage remained unstable. The diesel was stopped for further investigation by the system engineer. The cause of the problem was believed to be in the exciter bridge or voltage regulator, so the bridge circuit was switched from bridge No. 1 to bridge No. 2 and the diesel restarted. The surveillance was completed with no further problems. The cause was determined to be the remote gate firing module on bridge circuit No. 1. The inspector informed the licensee that the August 14, 1989 letter lists an incorrect failure date.

b. Deficiency Cards and Licensee Event Reports

Deficiency Cards and Licensee Event Reports were reviewed for potential generic impact, to detect trends, and to determine whether corrective actions appeared appropriate. Events which were reported pursuant to 10 CFR 50.72 were reviewed following occurrence to determine if the technical specifications and other regulatory requirements were satisfied. In-office review of LERs may result in further followup to verify that the stated corrective actions have been completed or to identify violations in addition to those described in the LER. Each LER was reviewed for enforcement action in accordance with 10 CFR Part 2, Appendix C, and where the violation was not cited, the criteria specified in Section V.G of the Enforcement Policy were satisfied. Review of DCs was performed to maintain a realtime status of deficiencies, determine regulatory compliance, follow the licensee corrective actions, and assist as a basis for closure of the LER when reviewed. Due to the numerous DCs processed, only those DCs which resulted in enforcement action or required further inspector followup with the licensee at the end of the inspection are listed below. The DCs and LERs denoted with an asterisk indicate that reactive inspection occurred following the event and prior to receipt of the written report.

(1) Deficiency Card reviews:

DC 2-89-1360, "Failure To Conduct Surveillance On 2-HV-15216D (Blowdown Isolation Valve)."

On September 3, 1989, it was discovered that valve 2HV-15216D (Blowdown Isolation Valve) had not been tested

prior to its absolute late date of August 29, 1989 at 1807 CDT. Operations Procedure 14825-2 Rev. 1 (Quarterly Inservice Valve Test) tests the valve pursuant to TS 4.0.5. Testing of this valve had been delayed while work was completed on an associated valve. Following completion of the work on the associated valve the test was not rescheduled. During final review of the work order, the licensee identified that the test was specified and had not been performed. The licensee determined that without the TS 4.0.5 required surveillance being completed, that the high energy line break instrument channel LCO requirements of TS 3.3.3.11 were not being satisfied. No action statement is provided when one of the two instruments are inoperable. With both instruments inoperable, the licensees' required action is to restore the inoperable instruments to an operable status within 7 days. Since an unlimited amount of time is allowed for a single inoperable instrument, the licensee determined that they were in compliance with the LCO at all times. The instrument became inoperable at the point in time the surveillance was not performed; therefore, the surveillance was not missed, since licensees are not required to perform surveillance on inoperable equipment. Thus, no reportable event had occurred. After discussion with NRR, the inspector informed the licensee that TS 3.3.3.11 requires all instrumentation to be operable, regardless of a lack of an action statement. Since the instrument was not known to be inoperable, a missed required surveillance had occurred and was reportable. This item will be further followed up when submitted as an LER.

(2) The following LERs were reviewed and are ready for closure pending verification that the licensee's corrective actions have been completed.

(a) 50-424/88-27, Rev. 0, "Procedure Inadequacy Leads To Containment Ventilation Isolation."

On October 4, 1988, a technician was in the process of changing out a circuit component in the Containment Low Range Radiation Monitor, 1RE-0003. When the monitor was placed in bypass and powered down, a CVI resulted. The cause of this event was an inadequate procedure that did not require the lifting of leads to the CVI actuation circuits before removing power. Revisions to procedure 24623-1 were to include the appropriate steps for the evolution. In addition, how to establish provisions for blocking and the conduct of a broadness review were to be performed.

Procedure revisions were verified by the the inspector. The broadness review indicated that no actions other than adding blocking switches could be implemented. The review for the blocking switches concluded that engineering would pursue a design change. The inspector requested to be provided the current status for design implementation. This LER remains open for review of the design change. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-424/88-51.

(b) 50-424/88-28, Rev. 1, "Safety Injection Initiated While Performing Test Procedure."

On October 16, 1988, with the Unit in Mode 6 (Refueling), an unplanned Safety Injection signal was generated, prior to the procedural step in which it was expected, while performing Procedure 54055-1 "Train "A" Diesel Generator And ESF System Actuation Test". The system engineer was aligning the Solid State Protection System to simulate a SI signal. While positioning the Logic B selector switch, a signal was generated which caused a SI actuation. The plant responded as expected with a SI signal present. There was no injection to the reactor vessel since the affected pumps were isolated from the Reactor Coolant System and on miniflow at the time. The cause of this event was personnel error. The system engineer failed to complete a prerequisite for the test. Corrective actions included an immediate change of procedures 54055-1 and 54065-1 to complete the ESF System Actuation Test using other circuitry and a rewrite of these procedures to make the procedures easier to follow. The rewritten procedures were reviewed by the inspector. However, during the review, the inspector highlighted numerous typographical errors. The inspector requested that the improvements between the old and new procedure be presented. The licensee responded to the request by having the new system engineer review the procedure. He determined that the procedure could be revised further. The procedure will be revised by the next refueling outage test. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-424/89-07.

(c) 50-424/88-37, Rev. 0, "O-Ring Found Missing In Post Accident Monitoring RTD's Junction Boxes."

On November 16, 1988, while performing Maintenance Work Order 18808056, O-rings were discovered to be missing from 4 CONAX T-8 Head junction boxes. Three of the boxes service resistance temperature detectors that provide

reactor coolant T-hot wide range temperature indication for post accident monitoring. The detectors were in an untested configuration. Technical Specification 3.3.3.6, "Accident Monitoring Instrumentation," requires that these detectors be operable during plant operation. On November 4, 1988, while reviewing environmental qualification documentation, it was noted that installation of O-rings was required in the tested configuration to seal the CONAX T-8 Head junction boxes. A check of material inventory revealed that no O-rings had been ordered as replacement spares. A MWO was written to inspect the subject boxes. During the inspection, four O-rings were discovered to be missing. A contributing factor to the occurrence of this event was the fact that the O-rings were not installed during initial installation. All the CONAX T-8 Head junction boxes were inspected under MWO 18808056. The missing O-rings were replaced and the boxes sealed. Environmental qualification documentation has been updated to specify the requirements for O-rings and maintenance procedures have been revised to address their replacement. Inclusive in this review was a Westinghouse review of the issue. The inspector questioned the comprehensiveness of the corrective actions since the environmental qualification package was different from the maintenance procedures. Pending resolution of these procedural inconsistencies, this LER remains open. This licensee identified violation is not being cited because criteria specified in Section V.G.1 of the NRC Enforcement Policy were satisfied.

NCV 50-424/89-25-02, "Failure To Maintain 10 CFR 50.49 Equipment Qualification On Three Temperature RTDs Due To Missing O-rings - LER 88-37".

(d) 50-425/89-25, Rev. 0, "Entry Into LCO 3.0.3 Due To Tripping Of ESF Room Coolers On Thermal Overload."

On August 1, 1989, conditions for entering Limiting Condition for Operation 3.0.3 occurred when a train "B" Engineered Safety Feature room cooler tripped on thermal overload. Shortly before this, a train "A" ESF room cooler had tripped on thermal overload. With a room cooler in each train of ESF out of service, a condition not provided for in the action requirements of Technical Specification 3.7.11 existed. Entry into LCO 3.0.3 was required. Several trips occurred for these room coolers on August 1. In each case, the trip was on thermal overload, and after each trip, the room cooler was restored to operable status by resetting the overload device and verifying the cooler fan was running. Subsequent to August 1, two other ESF

room coolers experienced trips on thermal overload. On August 3 and 4, after consultation with design engineering personnel, the heater coils installed in the thermal overload devices for room coolers 2-1555-A7-003, 2-1555-A7-015, and 2-1555-A7-016 were changed. This increased the trip settings of these overload relays. After the trip of room cooler 2-1555-A7-009 on August 3, it was decided to conservatively increase the thermal overload trip settings for all the Unit 2 and Unit 1 Auxiliary Building ESF room coolers to the maximum allowed by procedure. Additionally, it was decided to change out the entire thermal overload relay device for the room coolers which had experienced the tripping problems in order to perform testing to determine the root cause. The change out of the thermal overload relay devices and the change-out/repositioning of the overload heater coils for the Unit 2 Auxiliary Building ESF room coolers was completed by August 6. The changeout/repositioning of the overload heater coils for the Unit 1 Auxiliary Building ESF room coolers had been initiated but was not complete. This item does not represent a violation of NRC requirements.

(3) The following LERs were reviewed and closed.

(a) 50-424/87-80, Rev. 0, "Personnel Error Leads To Missed Secondary Coolant Sampling Surveillance."

Technical Specification Table 4.7-1 requires that the Secondary Coolant System specific activity be sampled and analyzed for gross radioactivity determination at least once per 72 hours when the unit is in Mode 1, 2, 3, or 4. On January 25, 1989, plant personnel were conducting a review of TS Surveillance records and were unable to locate evidence that documented the completion of this TS surveillance. The cause of this event was determined to be personnel error in failing to follow procedure 30025-C, "Periodic Analysis Scheduling Program." As a result of previous events, use of a TS scheduling board was initiated approximately 2 months prior to this event in 1987. Because there have been no other missed surveillances of this type in over 1-1/4 years, the board has been determined to be an adequate scheduling tool and this event represents an isolated incident. A new data sheet that was added to procedure 35210-C, "Chemistry Control Of The Steam Generators," was verified complete.

(b) 50-424/87-82, Rev. 0, "Failure To Perform Response Time Test Results In Technical Specification Violation."

On June 20, 1989, plant personnel were reviewing the maintenance history associated with the Unit 1 reactor trip

breakers when it was discovered that a RTB had been swapped-out without performing a response time test for the breaker being installed. This swap-out occurred on October 17, 1987, and involved the replacement of breaker serial No. 02YN072B-4 with breaker serial No. 860.759-1. This was in the main RTB "B" cubicle. Therefore, from October 17, 1987, until March 6, 1988, the minimum channels operable requirements of Technical Specification 3.3.1 were not met, as far as response time testing was concerned. The root cause of this event is considered to be procedural inadequacy in that Procedure 27765-C, "Westinghouse Type DS-416 Circuit Breaker Maintenance," did not contain instructions for performing a response time bench test. Procedure 27765-C, implements the RTB PM program, and this procedure has been changed to address response time testing requirements. The changes were verified to be included in the procedure. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-424/89-22.

(c) *50-424/88-31, Rev. 0, "Inadequate Procedure Leads To Plant Operation Outside Of Technical Specifications."

On October 30, 1988, core alterations were suspended while a Train "A" loss of offsite power test was conducted. While this occurred, the actuation of Train "A" Engineered Safety Features was blocked. When the test was completed, core alterations resumed. During shift change and turnover, control room personnel noticed that the protection system "Train "A" Trouble" annunciator was lit and discovered that the input error inhibit switch on the SSPS logic test panel was in the "inhibit" position. The CVI system is not operable unless both Train "A" SSPS and Train "B" SSPS are operable. Also, direct access to the outside atmosphere would not be blocked by closing the containment ventilation penetrations. Therefore, the plant was operating in a condition outside of the Technical Specifications and core alterations were halted. The input error inhibit switch was returned to it's "normal" position. SSPS train "A" was returned to operable status and core alterations were resumed. The cause of the event was due to the input error switch being moved to the "inhibit" position as part of a pretest lineup described in procedure 54055-1, "Train "A" Diesel Generator And ESFAS Test". This procedure did not provide for the restoration of this switch to the "normal" position and this procedural inadequacy was determined to be the cause of this event.

The procedure changes were reviewed by the inspector. During the review, the inspector noticed that inadequate change bars were used to reflect the changes made. This item resulted in a violation of NRC requirement, and was reported in NRC report 50-424/88-56.

(d) 50-424/88-32, Rev. 0, "Personnel Error Leads To Plant Operation Outside Of Technical Specification Requirements."

On November 4, 1988, during a documentation review, a setpoint discrepancy was identified. During refueling operations TS Table 3.3-3 requires a trip setpoint of 15 mrem/hr for area radiation monitors RE-0002 and RE-0003. On October 26, 1988, background subtract values were inserted into these monitors. This had the effect of raising the actuation setpoints to 17 mrem/hr for RE-0002 and 17.5 mrem/hr for RE-0003. This condition existed until November 3, when core alterations were completed and the normal trip setpoints were inserted. The cause of this event was personnel error due to procedure non-compliance in adding the background subtract values. The appropriate personnel were supposed to be counseled regarding strict compliance to procedure by December 15, 1988. This counselling did not occur until January 12, 1989. The inspector verified that the corrective actions were completed. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-424/88-56.

(e) 50-424/88-38, Rev. 0, "Erroneous Neutron Detector Indicators Lead To Plant Operation Outside Of Technical Specifications."

During the October 1987 maintenance outage, new software associated with the extended range neutron detectors 1NI-13135C and 1NI-13135D was installed as part of a Design Change Package. On October 18, 1988, an 18 month surveillance per Technical Specification 3.3.3.5.1, found the extended range neutron indicators on the remote shutdown panel to be indicating outside of the tolerance required. On November 16, 1988, while reviewing the deficiency card associated with the out-of-tolerance condition, the system engineer discovered that the indicators had been giving erroneous readings since the software change in October 1987. The erroneous signals were corrected prior to plant entry into Mode 3 (Hot Standby) and the individual responsible for review of the original DCP was counseled regarding this incident. The inspector verified that the corrective actions were completed. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-424/88-61.

(f) 50-424/88-40, Rev. 0, "Containment Ventilation Isolation Occurs During Calibration Of Radiation Monitors."

On November 21, 1988, plant personnel were conducting Technical Specification surveillance calibrations per procedure 43690C, "Calibration Of Area Monitors." While plant personnel were calibrating area radiation monitor 1RE-0003 it initiated a Containment Ventilation Isolation Signal. The appropriate valves and dampers actuated and control room operators verified that no abnormal radiation condition existed. The CVI signal was reset. The cause of the CVI was determined to be personnel error. Personnel calibrating area radiation monitor 1RE-0003 failed to verify that the data processing module was set in "bypass" before exposing the radiation monitor circuitry to a high radiation actuation signal. The concerned personnel were counseled regarding the importance of procedural compliance. The inspector reviewed the investigation.

(g) 50-424/88-42, Rev. 0, "Spurious High Radiation Alarm Leads To Containment Ventilation Isolation."

On December 14, 1988, a Containment Ventilation Isolation occurred due to a high radiation level alarm from the containment purge iodine monitor 1RE-2565B. The appropriate valves and dampers actuated. The control room operators verified that no abnormal radiation condition existed and the CVI signal was reset. An investigation confirmed that the monitor was registering normal background radiation levels at the time of the event and no cause could be found for the spurious high radiation actuation signal. The monitor was returned to service and closely monitored for a recurrence of this event. The inspector has no further questions.

(h) *50-424/88-45, Rev. 0, "Fuel Handling Building Isolation From High Radiation Caused By Personnel Error."

On December 29, 1988, while sampling the reactor coolant (with the Post Accident Sampling System reactor coolant line being backflushed), radiation monitors ARE 2532B and 2533B detected of gaseous activity of 7.7E-7 uCi/cc of Xe-133 in the Fuel Handling Building Heating, Ventilation, and Air Conditioning system. This caused an unplanned automatic Engineered Safety Feature actuation. The PASS had been tested on December 28 and isolation valve 1-HV-8220 had been inadvertently left open. This resulted

in a flow path for reactor coolant to discharge to FHB drains when the hot leg sample valve inside containment was opened while backflushing of the PASS was being performed. Corrective actions included: (1) re-emphasis of closed-loop communication for the Chemistry and Operations department personnel, (2) a caution to be added to the sections of 35611-C, Rev. 7 that perform a flush of the PASS panel to ensure 1-HV-8220 is closed, and (3) administrative control to maintain a closed door to room A-10. Corrective actions were verified complete. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-424/89-07.

(i) 50-424/89-08, Rev. 0, "Inadequate Review Of Drawing Change Results In Use Of Improper Breakers."

On February 23, 1989, it was discovered that 125 Vdc breakers for motor-operated valves in the Turbine Driven Auxiliary Feedwater pump system were not the proper size. The breakers, as installed and as shown on design drawings, were 15 amp thermal magnetic, but should have been sized as 30 amp thermal magnetic per the design criteria. Therefore, the plant operated in a condition prohibited by Technical Specifications. Technical Specification 3.7.1.2 requires at least three independent steam generator auxiliary feedwater pumps and flowpaths to be operable. The undersized breakers were discovered as a result of an investigation of the same problem in Unit 2. LCO 1-89-121 was entered. The breakers were replaced, successfully tested, and the LCO was exited. The cause of this event was due to inadequate review by the responsible engineer when a drawing change notice corrected the MOV horsepower rating from 0.66 hp to 1.0 hp. Corrective actions included a review of all 125 Vdc MOV breaker protection. This review indicated this incident to be an isolated case. The inspector verified the corrective actions. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-424/89-14.

(j) 50-424/89-14, Rev. 0, "Failure To Analyze Diesel Lube Oil Leads To License Condition Violation."

On June 2, 1989, it was discovered that the plant had not complied with the Operating License, Attachment 1, paragraph 5.c. Operating License NPF-68, Section 2.C(6), requires GPC to implement diesel generator requirements as specified in Attachment 1 to the license. Attachment 1, paragraph 5.c, mandates quarterly spectrographic and ferrographic analysis of engine oil to detect evidence of

bearing degradation. The quarterly ferrographic analysis had last been performed in October 1988, for the train "A" diesel generator. Additionally, spectrographic analyses had not been regularly trended to detect indication of abnormal bearing degradation. The cause of this event was the failure to adequately incorporate license commitments into plant procedures. A ferrographic analyses was performed and found acceptable based on comparison with previously taken baseline data. Corrective actions included revision to Procedures 54170-1, "Diesel Generator Lube Oil Analysis, Trending And Evaluation," to require trend evaluation of quarterly spectrographic and ferrographic analyses, and to Procedure 32531-C, "Diesel Generator Lube Oil Sampling And Analysis," to require engine oil samples to be taken for ferrographic analysis. The procedure changes were reviewed by the inspector. During the review, the inspector noted that procedure 54170-C did not include change bars where changes were made. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-424/89-19.

(k) 50-424/89-16, Rev. 1, "Manual Reactor Trip Due To Failure Of Main Feedwater Isolation Valve."

On July 8, 1989, a manually initiated reactor trip occurred on Unit 1 from at 100% of rated thermal power. The manual trip was initiated because the Loop 4 Main Feedwater Isolation Valve failed closed resulting in a decrease in the number 4 Steam Generator level. The plant was stabilized in Mode 3 following the reactor trip. Troubleshooting following the manual trip failed to identify the exact cause of the spurious closure; however, two potential failure mechanisms were evaluated - solenoid failure and auxiliary relay failure. Corrective actions for the July 8 event included troubleshooting of the valve control circuitry, repair of the associated handswitch for a problem noted during troubleshooting but unrelated to the failure, and continued control loop monitoring with a multi-channel recorder. On August 3, the Loop 4 MFIV spuriously closed. The resulting decrease in feedwater led the operator to trip the reactor. Corrective action for this event consisted of replacing a failed solenoid valve and returning the valve to the vendor for failure analysis. The inspector has no further questions.

(l) *50-424/89-17, Rev. 0, "Radiation Monitors High Alarm Results In Fuel Handling Building Isolation."

On July 28, 1989, with the unit at 100% power, radiation monitor ARE-2532B went to its high alarm setpoint, causing

a Fuel Handling Building Isolation. The Post Accident Filtration units started and the appropriate vents and dampers actuated. Control room operators announced the isolation over the public address system and began to survey other area monitors to check for the possible spread of radioactive contaminants. No other monitor was found with an abnormal reading. The activity level on ARE-2532B dropped below the high alarm setpoint and the isolation signal was reset. The root cause of this event was not determined after conducting a team investigation. To preclude recurrence, the licensee has raised the alarm setpoint. The inspector has no further questions.

(m) 50-425/89-05, Rev. 0, "Inadequate Review Of A Modification Results In A Technical Specification Violation."

On March 17, 1989, while investigating a problem with the Automatic Surveillance Technical system, field voltage measurements were taken that revealed an electrical short on valve 2HV-19051, the Reactor Coolant Pump #1 thermal barrier isolation valve. The valve was required to be operable upon entry into Mode 4, which had occurred on March 4. A Surveillance had been performed on February 4, to prove operability of 2HV-19051; however, a change to the system wiring on February 10 resulted in valve 2HV-19051 being made inoperable. The cause of this event was the issuance of an incorrect As-Built-Notice. Corrective actions included counseling the appropriate engineering personnel involved, training for all engineering personnel recently transferred from the Unit 2 test organization on use of the ABN, and issuing a second ABN to restore the system to its original configuration. The inspector verified the corrective actions were completed. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-425/89-15.

(n) *50-425/89-06, Rev. 0, "Operation Of Incorrect Handswitch Results In Safety Injection."

On March 18, 1989, while warming the main steam lines as part of Procedure 12002-2, "Unit Heatup To Normal Operating Temperature And Pressure," an automatic Engineered Safety Features actuation occurred. A step of the procedure called for handswitches HS 40047/48 to be operated to reset the main steam isolation signal. However, handswitches HS 40068/69 were operated. These switches reset the low steamline pressure safety injection and steamline isolation logic, removing the blocking signal. Since the main steam

line pressure was below the safety injection setpoint pressure, the SI occurred. Appropriate ECCS pumps and valves actuated resulting in approximately 2900 gallons being injected into the Reactor Coolant System. The SI was manually reset and injection into the RCS was terminated. The cause of this event was personnel error. The operator failed to ensure that the proper switch was being operated. Corrective actions included counseling the operator on the importance of verifying that proper devices are operated, changing the color of SI handswitches, adding cautions to the handswitches, and incorporating details of this event into training. This item was formally discussed following an enforcement conference on March 22. The inspector has verified that the corrective action have been completed. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-425/89-15.

(o) 50-425/89-07, Rev. 0, "Lockup Of A Computer Communications Device Results In Containment Ventilation Isolation."

On March 19, 1989, while restoring the Plant Effluent Monitoring System to service the plant experienced an automatic Engineered Safety Features actuation which resulted in a Containment Ventilation Isolation. Appropriate valves and dampers actuated to isolate containment ventilation. Control room operators verified that no abnormal radiological conditions existed using 2RE-0002/0003. The monitor that actuated the CVI, 2RE-2565, was placed in bypass. The CVI was reset and equipment that actuated was returned to normal operating position. Due to an earlier SI, power had been lost to most of the PERMS system. On restoration of power, the computer parameter files were initialized with a -9.99E-20 value. (The computer replaces this value with parameters received from each monitor.) Due to a communication failure of a multiplexer, communication with several monitors was lost and no value was received for 2RE-2565. When the mutiplexer was reset, the computer detected the original power failure for 2RE-2565. The computer gave the monitor the current parameter on file and assigned the monitor -9.99E-20 value. This resulted in a high alarm, causing the CVI actuation. Procedure changes were reviewed by the inspector and corrective actions were verified. The inspector has no further questions.

(p) 50-425/89-12, Rev. 0, "Operating Incorrect Switch Results In Inoperable Monitor Requiring Entry Into TS 3.0.3."

On March 30, 1989, while performing maintenance on 2RE-2562A, an Instrument and Controls Technician inadvertently placed 2RE-2562A and 2RE-2562C in purge instead of activating the paper drive on 2RE-2562A. This caused 2RE-2562C to become inoperable. Later the same day, a chemistry foreman discovered 2RE-2562C to be inoperable and notified the control room. An entry into TS 3.0.3 was made due to an existing Limiting Condition for Operation for the Reactor Coolant System Leakage Detection System and 2RE-2562C being inoperable. With 2RE-2562C inoperable the LCO for Technical Specification 3.4.6.1 could not be met. 2RE-2562C was restored to service and TS 3.0.3 exited. The cause of this event was personnel error. The I & C technician failed to pay attention to detail when activating plant equipment. The purge switch was activated instead of the paper drive. Corrective actions included counseling the individual and issuing a memo to all I & C personnel concerning attention to detail when performing maintenance/trouble shooting on plant equipment. The inspector has verified that the corrective actions were completed. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-425/89-15.

(q) *50-425/89-16, Rev. 0, "Unplanned Auxiliary Feedwater Actuation On Recovery From Remote Shutdown Test."

On April 11, 1989, while recovering from a Remote Shutdown Test, an automatic Engineered Safety Features actuation (auto start signal to motor driven Auxiliary Feedwater pumps) occurred. During the Remote Shutdown test, both Main Feedwater Pumps were manually tripped and AFW was in service. With both MFPs tripped, an AFW actuation signal was generated; however, with control at the Remote Shutdown Panel, the signal was interrupted. When control was returned to the control room, the signal was reinstated. As the AFW pumps were already in operation, the AFW actuation signal caused the discharge valves of the Train "A" to stroke full open. Control room operators immediately throttled AFW flow to prevent overfilling of the steam generators. MFP "A" was reset to allow return of the remaining trains to the control room. All AFW systems were restored to readiness. The cause of this event was a situation that was not anticipated by the procedure. Procedure 18038-2, "Operation From Remote Shutdown Panels," was to be revised to caution operators of the possible actuation upon transfer of control to the control room. Procedure changes were reviewed by the inspector and corrective actions verified.

(r) 50-425/89-17, Rev. 0, "Inadequate Procedure Leads To Missed Technical Specification Surveillance."

On April 15, 1989, during startup of the reactor, gas was vented from the Pressurizer Relief Tank to Waste Gas decay tank #10 to clear the high pressure alarm for the PRT. The Radwaste Operator and Unit 2 Control Room personnel failed to notify Chemistry of the transfer. Therefore, the waste gas tank was not sampled within 24 hours as required by Technical Specification Surveillance 4.11.2.6. On April 18, during a routine walkdown inspection, the Radwaste Supervisor noted an increase in pressure in waste gas decay tank #10 and notified Chemistry. A review of Chemistry records did not indicate that any additions had been made to waste gas decay tank #10. As normal sampling could not be performed due to an inadequate tank pressure, a standing order for the collection of local samples was developed and the tank was sampled on April 20. The cause of this event was a procedure that was less than adequate. The steps of procedure 13201-2, "Gaseous Waste Processing System" that address venting of the PRT did not require the notification of Chemistry. Procedure 13201-2 (13201-1 for Unit 1) were revised to include a caution to notify Chemistry of any change in system status and any addition or transfer of waste gas within the system. This action was completed on July 27, 1989. The inspector verified the corrective actions. This item resulted in a violation of NRC requirements and was reported in NRC Inspection Report No. 50-425/89-18.

(s) *50-425/89-23, Rev. 0, "ESF Actuation Results When Transferring Offsite Power Sources."

On July 20, 1989, with the unit at 100% power, a non-1E 4160 volt bus, 2NA05, was being transferred from the Reserve Auxiliary Transformer to the Unit Auxiliary Transformer, when RAT 2A tripped on a differential relay lockout. Power was lost to the Train "A" 1E bus and various non-1E buses. The loss of power resulted in a diesel generator start and actuation of the Auxiliary Feedwater system. An investigation found that a current transformer in the 2NA05 4160V bus had been improperly terminated. The improper termination had gone undetected because the normally small load carried by the bus was insufficient to actuate the differential lockout relay under normal conditions. However, during the transfer, the momentary surge of current, as both the normal and alternate supply breakers were closed, was sufficient to actuate the relay. The cause of this event was an error in

the original design. Corrective action included correcting the improper termination and reviewing other breakers in both units for similar wiring conditions. Corrective actions were reviewed by the inspector as part of event followup.

(t) *50-425/89-24, Rev. 0, "Failure Of Pressure Channel Circuit Card Causes Reactor Trip."

On July 26, 1989, with the unit at 100% power, a 2 out of 4 OT delta-T trip signal was received and caused an automatic reactor trip. At the time, personnel were performing corrective maintenance on power range nuclear instrumentation channel 2N43. The corrective maintenance on 2N43 required the channel III reactor trip bistable to be tripped as a part of the removal from service process. A loss of input from pressurizer channel 2P458 then occurred. This caused the channel IV reactor trip bistable to trip and completed the 2 out of 4 logic required for reactor trip. The failure of channel 2P458 was caused by the failure of an operational amplifier in the non-isolated section of an NLP2 process card. This channel had spiked low on two separate occasions several days earlier but troubleshooting had failed to identify the exact cause of the problem. An additional spiking problem had been experienced on channel 2N43 and was still being investigated at the time of the event. Corrective action consisted of replacing the defective NLP2 card for channel 2P45 and replacing a suspect card for channel 2N43. Corrective actions were monitored by the inspector as part of event followup.

One non-cited violation was identified in paragraph 3.b(2)(c).

4. Actions on Previous Inspection Findings - (92701)(92702)

(Closed) UNR 50-424/88-16-01, "Review Single Mode Failure For The Volume Control Tank Outlet Valves."

This item concerned the effect of a fire induced spurious closure of a VCT outlet valve (LV-112B or LV-0112C) and subsequent failure of the centrifugal charging pumps in this condition. To preclude this failure mode, the licensee executes a pump suction swap-over to the Refueling Water Storage Tank before abandoning the main control room. Since licensees are evaluated during the inspection on the basis of only tripping the reactor prior to control room abandonment with all spurious failures occurring, this failure mode could affect the safe shutdown of the unit. In a NRC memorandum dated December 6, 1988, the Plant Systems

Branch of NRR concluded that the Vogtle design was acceptable and that, due to the generic implications of the issue, an information notice describing the scenario would be developed. The notice will alert other licensees to the possible need to revise their procedures regarding operator action from the control room prior to abandonment. Since the Vogtle design was determined acceptable and procedures are in place, this item is closed and no NRC violations are apparent.

5. Exit Interviews - (30703)

The inspection scope and findings were summarized on September 15, 1989, with those persons indicated in paragraph 1 above. The inspectors described the areas inspected and discussed in detail the inspection results. No dissenting comments were received from the licensee. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspector during this inspection. Region based NRC exit interviews were attended during the inspection period by a resident inspector. This inspection closed one Unresolved Item (paragraph 4) and twenty Licensee Event Reports (paragraph 3.b(3)). The items identified during this inspection were:

VIOLATION 50-424/89-25-01 and 50-425/89-29-01, "Failure to Adequately Certify Pursuant to 10 CFR 55.23 that a Medical Examination Had Been Performed Which Met 10 CFR 55.21 Requirements." - paragraph 2.b(9).

NCV 50-424/89-25-02, "Failure To Maintain 10 CFR 50.49 Equipment Qualification On Three Temperature RTDs Due To Missing O-rings - LER 88-27." - paragraph 3.b(2)(c).

The inspector informed the licensee that the violations identified above may be characterized differently during the inspection report approval process. Depending on the severity of the violation or to be consistent with recently identified problems in operator qualifications the issuance of Notice may be delayed or an Unresolved Item may result. The inspector confirmed that the licensee was committed to completion of the corrective actions.

The inspector also informed the licensee that, due to utility initiative, one facility tour of the turbine building was conducted with the On Shift Operations Supervisor. During this tour, the inspector was informed that the plant areas had recently been divided among the other supervisors as an operations department initiative. The inspector thanked the licensee for cooperation during the NRC sponsored tour of the two Yugoslavian visitors. The visitors noted that this plant was at a higher level of cleanliness and more health physic restrictive on personnel movement than Yugoslavian plants. They were appreciative of the plant photos. The information provided to the inspectors on the licensee plans to prevent losses of operators and provide career opportunities was useful in reducing concern that talented personnel were being lost to other industries.

In summary, the inspector identified that this inspection noted one strength, in the area of Radiation Controls, regarding the process of performing investigations (paragraph 2.b(4)). Weaknesses were identified in the technical support area regarding the medical qualification process of licensed operators (paragraph 2.b(9)) and in the area of plant operations regarding the updating, completeness, and use of qualification lists by the Shift Supervisors (paragraph 2.a).

6. Acronyms and Initialisms

ABN	As Built Notice
ACOT	Analog Channel Operability Test
AFW	Auxiliary Feedwater System
ANSI	American National Standard Institute
CC	Cubic Centimeter
CDT	Central Daylight Time
CFR	Code of Federal Regulations
CONAX	(trade name)
CS	Containment Spray System
Cs	Cesium
CVCS	Chemical & Volume Control System
CVI	Containment Ventilation Isolation
DC	Deficiency Cards
DCP	Design Change Package
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Features
ESFAS	Engineering Safety Features Actuation System
EST	Eastern Standard Time
FHB	Fuel Handling Building
GPC	Georgia Power Company
hp	Horsepower
hr	Hour
HS	Handswitch
HV	High Voltage
I&C	Instrumentation and Controls
KVAR	Kilovolt Amps Reactive
KW	Kilowatt
LCO	Limiting Conditions for Operations
LER	Licensee Event Reports
MDD	Minor Departure from Design
MFIV	Main Feedwater Isolation Valve
MFP	Main Feed Pump
MOV	Motor Operated Valve
mrem	Millirem
MWO	Maintenance Work Order
nCi	NanoCurie
NCV	Non-cited Violation

NI	Nuclear Instrument
NIS	Nuclear Instrument System
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSCW	Nuclear Service Cooling Water System
OT	Over Temperature
PASS	Post Accident Sampling System
PERMS	Process Effluent Radiation Monitor System
PM	Planned Maintenance
PRT	Pressurizer Relief Tank
RAT	Reserve Auxiliary Transformer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
Rev	Revision
RHR	Residual Heat Removal System
RTB	Reactor Trip Breaker
RTD	Resistance Temperature Detector
SI	Safety Injection System
SSPS	Solid State Protection System
Tave	Reactor Coolant System Average Temperature
TDAFW	Turbine Driven AFW Pump
TS	Technical Specification
uCi	Micro Curie
UNR	Unresolved Item
VCT	Volume Control Tank
Vdc	Volts direct current
Xe	Xenon