4 SURVEILLANCE REQUIREMENTS

4.0 SURVEILLANCE STANDARDS

Applicability

Applies to surveillance requirements which relate to tests, calibrations and inspections necessary to assure that the quality of structures, systems and components is maintained and that operation is within the safety limits and limiting conditions for operation.

Objective

To specify minimum acceptable surveillance requirements.

Specification

4.0.1 Surveillance of structures, systems, components and parameters shall be as specified in the various subsections to this Technical Specification section, Section 4.0, except as permitted by Technical Specifications 4.0.2 and 4.0.3 below.

*4.0.2 Minimum surveillance frequencies, unless specified otherwise, may be adjusted as follows to facilitate test scheduling:

Specified Frequency	Maximum Allowable Interval Between Surveillances
Five times per week	2 days
Two times per week	5 days
Weekly	10 days
Bi-Weekly	20 days
Monthly	45 days
Bi-Monthly	90 days
Quarterly	135 days
Semiannually	270 days
Annually	18 months
Refueling Outage	22 months, 15 days

4.0.3 If conditions exist such that surveillance of an item is not necessary to assure that operation is within the safety limits and limiting conditions for operation, surveillance need not be performed if such conditions continue for a length of time greater than the specified surveillance interval. Surveillance waived as a result of this specification shall be performed prior to returning to conditions for which the surveillance is necessary to assure that operation is within safety limits and limiting conditions for operation.

*The requirements of specification 4.0.2 for surveillances performed on a refueling outage schedule are waived for Unit 1 until 11:59 p.m. July 15, 1983.

4.0.4 Inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components.

b. Has a duration sufficient to establish accurately the change in leakage between the Type A test and the supplemental test.

c. Requires the quantity of gas bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total leakage rate at Pa (59 psig) or P_t (29.5 psig).

4.4.1.1.5 Report of Test Results

The results of periodic tests shall be the subject of a summary technical report which shall be submitted to the Commission within 90 days of completion of the test.

4.4.1.2 Local Leak Rate Testing

4.4.1.2.1 Scope of Testing

The local leak rate shall be measured for the components listed in Table 4.4-1 in accordance with the criteria specified in Appendix J of 10CFR50.

*4.4.1.2.2 Frequency of Test

Local leak rate tests shall be conducted with gas at a pressure of not less than 59 psig during each reactor shutdown for refueling or other convenient interval but in no case at intervals greater than 24 months.

4.4.1.2.3 Acceptance Criteria

The combined leakage rate from all penetrations and isolation valves shall not exceed 0.125 weight percent of the postulated post-accident containment air mass per 24 hours at 59 psig.

4.4.1.3 Reactor Building Modifications

Any major modification or replacement of components affecting the Reactor Building integrity shall be followed by either an integrated leak rate test or a locak leak rate test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.3 and 4.4.1.2.3, respectively.

4.4.1.4 Isolation Valve Functional Tests

Inservice testing of ASME Code Class 1, 2, and 3 valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components.

*The requirements of section 4.4.1.2.2 are waived for the electrical and mechanical penetrations in Unit 1 until 11:59 p.m. July 15, 1983.

4.18 SNUBBERS

Applicability

Applies to hydraulic and mechanical snubbers used to protect the Reactor Coolant System and other safety-related systems.

Objective

To verify that the required hydraulic and mechanical snubbers are operable.

Specification

*4.18.1 Each snubber associated with the Reactor Coolant System and other safety-related systems, as specified in the appropriate Station Procedure shall be visually inspected. Visual inspections shall verify:

(1) that there are no visible indications of damage or impaired OPERABILITY,

(2) attachments to the foundation or supporting structure are secure, and

(3) in those locations where mechanical snubber movement can be manually induced, the snubbers shall be inspected as follows:

(a) At each refueling, the inaccesible snubbers shall be inspected near the beginning and the end of the outage.

(b) In the event of a severe dynamic event, snubbers in that system which experienced the event shall be inspected during the refueling outage to assure that the snubbers have freedom of movement and are not frozen up. The inspection shall consist of verifying freedom of motion using one of the following: (i) Manually induced snubber movement, (ii) evaluation of in place snubber piston setting; (iii) stroking the mechanical snubber through its full range of travel. If one or more mechanical snubbers are found to be frozen up during this inspection, those snubbers shall be replaced (or overhauled) before returning to power. Re-inspection shall subsequently be performed according to the schedule listed below.

Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.18.4. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be tested

*The inspection period requirements of Section 4.18.1 are waived for the inaccessible mechanical snubbers on Unit 1 until 11:59 p.m. July 15, 1983.

Attachment 2

Discussion of Proposed Technical Specification Revision

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

1) Section 4.4.1.2.2; Frequency of Test for the Local Leak Rate Testing.

2) In accordance with the criteria specified in Appendix J of 10CFR50.

(B) Structure/System/Component/Parameter Effected:

Electrical Penetration O-ring Seal (penetration number 62 of Table 4.4-1).

(C) Test/Calibration/Inspection Procedure name:

Electrical Penetration O-Ring Seal Leak Test

(D) Test/Calibration/Inspection Procedure Number:

PT/0/A/150/5

(E) Test/Calibration/Inspection Function:

To determine the leak rate through the double o-ring seal between electrical penetrations and Reactor Building flanges.

II. Bases for Requesting Relief

(A) Date procedure last performed:

The electrical penetration o-ring seal leak test for Unit 1 began on May 15, 1981 and was completed by June 30, 1981.

(B) Date Due:

May 15, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 30 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.4.1.2.2 requires that a local leak rate test be conducted during each Reactor Shutdown for refueling or some other convenient interval, but in no case at intervals greater than 24 months.

III. Justification for Relief

A review of all past electrical penetration o-ring seal tests was performed and a total of 4 failures have been identified. Only two are associated with failure of the o-ring seal. These failures were found March 6 and 18, 1980 in Unit 2. The last leak rate test conducted on Unit 1's electrical penetration revealed only one failure. However, the failure was not associated with the o-ring seal, but with the quick disconnect fitting that is used to connect the testing equipment to perform the test.

Penetration rooms are formed adjacent to the outside surface of each Reactor Building by enclosing the area around the majority of the penetrations. Each unit 1's penetration room is provided with two fans and two filter assemblies. Both fans, discharging through a single line to the unit vent, are controlled from the main control room. During normal operation, this system is held on standby with each fan aligned with a filter assembly. The engineered safeguards signal from the Reactor Building will actuate the fans.

Particulate filtration is achieved by a medium efficiency pre-filter and a high efficiency (HEPA) filter. Absorption filtration is accomplished by an activated charcoal filter. When the system is in operation, a negative pressure will be maintained in the penetration room to assure inleakage. Penetration room pressure is displayed in the control room and excessive and insufficient vacuum are annunciated. It can be assumed that no pressure differentials exist in the room, so that an instrument string sensing pressure at a single point can be used. This is because the communicative paths between various parts of the penetration room are very large in comparison with the minute leakage that might exist due to imperfect seals.

From time to time the system will be activated to purge the filters of any moisture that may accumulate. The air will be taken from the penetration room where it will be sufficiently warm to accomplish this purpose. Dampers are placed in the system inlets to prevent moisture being carried through by natural circulation.

The barrier leakage in the Reactor Building is the one-quarter inch steel liner plate. All penetrations are continuously welded to the liner plate before the concrete in which they are embedded is placed. The penetrations, shown in Figure 1 become an integral part of the liner and are so designed, installed, and tested.

The steel liner plate is securely attached to the prestressed concrete Reactor Building and is an integral part of the structure. The Reactor Building is conservatively designed and rigorously analyzed for the extreme loading conditions of a highly improbable hypothetical accident, as well as for all other types of loading conditions which could be experienced.

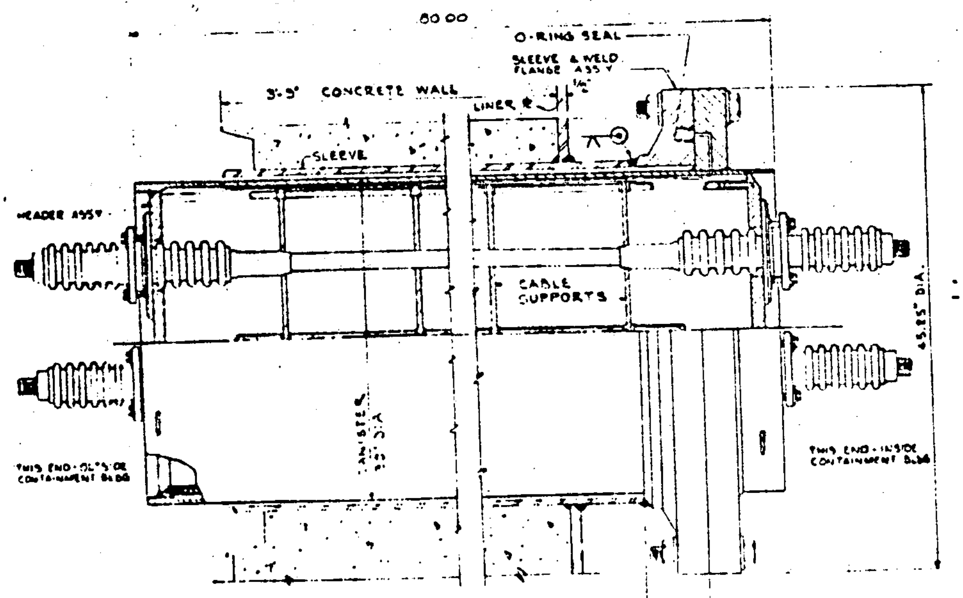
Under all normal operating conditions and under accidental conditions short of the worst loss-of-coolant accident, virtually no possibility exists that any leakage could occur or that the integrity of the vapor barrier could be violated in any way that would be significant to the public health and safety or to that of the station personnel. Adequate administrative controls will be enforced to minimize the possibility of human error.

All electrical penetrations are grouped within or vented to the penetration room. Any leakage that might occur from these penetrations will be collected and discharged through high efficiency particulate air (HEPA) filters and charcoal filters to the unit vent.

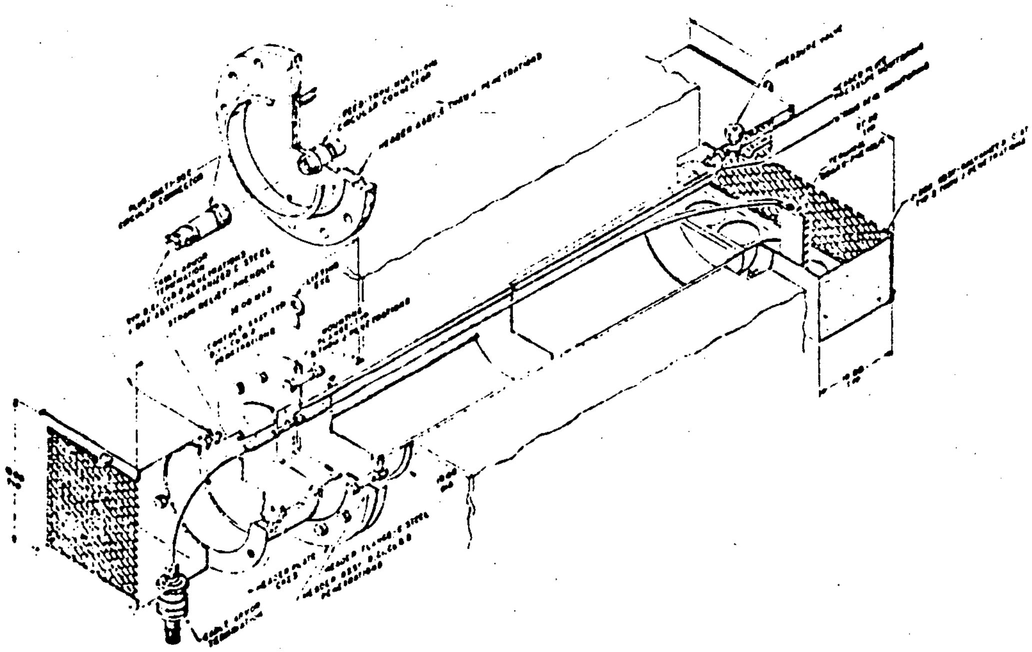
In conclusion, Duke Power has evaluated the significance of failure of the electrical penetration seals. Any leakage that would take place would be collected and processed by the Reactor Building Penetration Room Ventilation System (PRVS). The PRVS is designed to minimize environmental activity levels resulting from post-accident Reactor Building leaks.

The design basis for filtration was a requirement to remove 25 percent of the core iodine inventory. The 25 percent was derived using the standard assumption that during Maximum Hypothetical Accident 50 percent of the halogens are released from the core and that 50 percent of the iodine released plates out within the Reactor Building.

Finally, an operating experience review has shown failure of the o-ring seals are relatively low. More than a thousand leak rate tests have been performed and only two failures of o-ring seals have been found.



TYPICAL MEDIUM VOLTAGE POWER 6900 VOLT PENETRATION



TYPICAL LOW VOLTAGE POWER CONTROL & INSTRUMENTATION PENETRATION
NO SCALE

Figure 1

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

1) Section 4.4.1.2.2; Frequency of Test for the Local Leak Rate Testing.

2) In accordance with the criteria specified in Appendix J of 10CFR50.

3) ASME Section XI inservice inspection code, Subsection IWV.

(B) Structure/System/Component/Parameter Effected:

Type C Mechanical Penetrations (per table 4.4-1).

(C) Test/Calibration/Inspection Procedure name:

Local Type C Leak Rate Test

(D) Test/Calibration/Inspection Procedure Number:

PT/1/A/150/6

(E) Test/Calibration/Inspection Function:

To determine the leak rates of the Reactor Building Containment Isolation valves in accordance with the Technical Specifications, 10CFR50 Appendix J, and ASME Section XI, Subsection IWV.

II. Bases for Requesting Relief

(A) Date procedure last performed:

The mechanical penetration leak rate test for Unit 1 began on July 4, 1981 and was completed by August 10, 1981.

(B) Date Due:

July 4, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have been shutdown for refueling by the due date, if there are no forced outages until refueling.

(D) Required Surveillance Frequency:

Technical Specification 4.4.1.2.2 requires that a local leak rate test be conducted during each reactor shutdown for refueling or some other convenient interval, but in no case, at intervals greater than 24 months.

III. Justification for Relief

The barrier to leakage in the Reactor Building is the one-quarter inch steel liner plate. All penetrations are continuously welded to the liner plate before the concrete in which they are embedded is placed. The penetrations, become an integral part of the liner and are so designed installed, and tested.

The steel liner plate is securely attached to the prestressed concrete Reactor Building and is an integral part of the structure. The Reactor Building is conservatively designed and rigorously analyzed for the extreme loading conditions of a highly improbable hypothetical accident, as well as for all other types of loading conditions which could be experienced.

Under all normal operating conditions and under accident conditions short of the worst loss-of-coolant accident, virtually no possibility exists that any leakage could occur or that the integrity of the vapor barrier could be violated in any way that would be significant to the public health and safety or to that of the station personnel. Adequate administrative controls will be enforced to minimize the possibility of human error.

All mechanical penetrations are grouped within or vented to the penetration room. Any leakage that might occur from these penetrations will be collected and discharged through high efficiency particular air (HEPA) filters and charcoal filters to the unit vent.

Penetration rooms are formed adjacent to the outside surface of each Reactor Building by enclosing the area around the majroity of the penetrations. Unit 1's penetration room is provided with two fans and two filter assemblies. Both fans, discharging through a single line to the unit vent, are controlled from the main control room. During normal operation, this system is held on standby with each fan aligned with a filter assembly. The engineered safeguards signal from the Reactor Building will actuate the fans.

Particulate filtration is achieved by a medium efficiency pre-filter and a high efficiency (HEPA) filter. Absorption filtration is accomplished by an activated charcoal filter. When the system is in operation, a negative pressure will be maintained in the penetration room to assure inleakage. Penetration room pressure is displayed in the control room and excessive and insufficient vacuum are annunciated. It can be assumed that no pressure differentials exist in the room, so that an instrument string sensing pressure at a single point can be used. This is because the communicative paths between various parts of the penetration room are very large in comparison with the minute leakage that might exist due to imperfect seals.

From time to time the system will be activated to purge the filters of any moisture that may accumulate. The air will be taken from the penetration room where it will be sufficiently warm to accomplish this purpose. Dampers are placed in the system inlets to prevent moisture being carried through by natural circulation.

In summary, Duke Power has evaluated the significance of failure of these penetrations. Any leakage that would take place would be collected and processed by the Reactor Building Penetration Room Ventilation System (PRVS). The PRVS is designed to minimize environmental activity levels resulting from post-accident Reactor Building leaks.

The design basis for filtration was a requirement to remove 25 percent of the core iodine inventory. The 25 percent was derived using the standard assumption that during a Maximum Hypothetical Accident 50 percent of the halogens are released from the core and that 50 percent of the iodine released plates out within the Reactor Building.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

1) Section 4.18.1 - The visual inspection period to verify snubber operability.

(B) Structure/System/Component/Parameter Effected:

The mechanical snubbers on the Steam Generator Flush and Drain Line.

(C) Test/Calibration/Inspection Procedure name:

Inspection of inaccessible mechanical snubbers.

(D) Test/Calibration/Inspection Procedure Number:

MP/1/A/3018/19

(E) Test/Calibration/Inspection Function:

To visually inspect inaccessible mechanical snubbers to ensure that they are operable and no damage has occurred.

II. Bases for Requesting Relief

(A) Date Procedure last performed:

This inspection of the inaccessible mechanical snubbers on Unit 1 were completed on July 6, 1981.

(B) Date Due:

May 26, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 25 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specifications 4.18.1 requires that the maximum allowable interval between visual inspection be 22 months, 15 days.

III. Justification for Relief

Five Unit 1 pipe supports (04A-0-478A-NPS-H35, H35A, H37, H54 and NPS-04A-0-479-H41) that contain mechanical snubbers were inspected during Unit 1's 1981 refueling and 10 year inservice inspection outage. An evaluation for the effects on system operability if these snubbers were assumed to be inoperable was performed.

Based on this evaluation, it was determined that system operability is assured if supports H35, H35A and H37 were to be inoperable, due to the low stress conditions that exist in the piping and the reserve capacities that adjacent pipe supports possess. Support H59 is required to assure system operability; however, this support had a new snubber installed near the end of the Unit 1 outage per the requirements of IEB 79-14. Follow-up discussions confirmed that this support was installed and stroke tested in October, 1981. Thus, this supports next required inspection would be August, 1983. Support H41 was determined to be located on non-safety piping. This support was installed as part of SMR 31S. Subsequent review of the adjacent safety related piping for IEB 79-14 and NSM 1012-3108 did not require any seismic supports on this portion of the non-safety piping. H41 has no effect on the operability of either the safety or non-safety piping and can thus be left in place or removed.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.6.4 - Simulated Emergency transfer of the 4160 volt main buses to startup transformer and to the 4160 standby buses be performed during refueling outages.

(B) Structure/System/Component/Parameter Effected:

Emergency Power circuitry to switch power from standby buses to startup sources.

(C) Test/Calibration/Inspection Procedure name:

Emergency Power Switching Logic Standby Breaker Closure Channel A&B

(D) Test/Calibration/Inspection Procedure Number:

PT/1/A/610/1H

(E) Test/Calibration/Inspection Function:

To verify the circuitry utilized to transfer to the standby bus and retransfer to the startup source.

II. Bases for Requesting Relief

(A) Date procedure last performed:

September 18, 1981

(B) Date Due:

July 6, 1983

(C) Plant Status:

It is anticipated that Unit 1 would have shutdown for refueling by the due date, if there were no forced outages until refueling.

(D) Required Surveillance Frequency:

Technical Specification Section 4.0.2 requires that the maximum allowable interval between surveillance be 22 months, 15 days.

III. Justification for Relief

The anticipated shutdown date for Unit 1 (7/3/83) is prior to the due date (7/6/83) of this test. Thus there is no requirement to perform this test prior to the refueling outage. If there are unanticipated forced outages which delay the refueling shutdown date and if the forced outages are of insufficient length in order to perform this test, the situation requiring performance of this test prior to shutdown for refueling will exist. However, if there is a forced outage of sufficient length this test will be performed, thus fulfilling our surveillance requirements.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 10.

(B) Structure/System/Component/Parameter Effected:

Reactor Protection System, reactor coolant flow instrumentation and associated circuitry.

(C) Test/Calibration/Inspection Procedure Name:

Reactor Protection System Channel A Reactor Coolant Flow Instrument Calibration.

(D) Test/Calibration/Inspection Procedure Number:

IP/1/A/305/1I

(E) Test/Calibration/Inspection Function:

To test Integrated String for proper calibration and to calibrate analog and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 16, 1981

(B) Date Due:

May 2, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 45 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. It was found that the acceptance criteria for the Loop A reactor coolant flow instrumentation was exceeded. However, the acceptance criteria for the total flow was met. The reactor protective system uses the total flow in its calculations.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 10.

(B) Structure/System/Component/Parameter Effected:

Reactor Protection System, reactor coolant flow instrumentation and associated circuitry.

(C) Test/Calibration/Inspection Procedure Name:

Reactor Protection System Channel B Reactor Coolant Flow Instrument Calibration.

(D) Test/Calibration/Inspection Procedure Number:

IP/1/A/305/1J

(E) Test/Calibration/Inspection Function:

To test Integrated String for proper calibration and to calibrate analog and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 19, 1981

(B) Date Due:

May 5, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 45 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was not met due to a blown transmitter. The trainsmitter was replaced and calibrated. The calibration performed December, 1979, was investigated and the acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. <u>Surveillance Requirements for which Relief is Requested</u>

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 10.

(B) Structure/System/Component/Parameter Effected:

Reactor Protection System, reactor coolant flow instrumentation and associated circuitry.

(C) Test/Calibration/Inspection Procedure Name:

Reactor Protection System Channel C Reactor Coolant Flow Instrument Calibration.

(D) Test/Calibration/Inspection Procedure Number:

IP/1/A/305/1K

(E) Test/Calibration/Inspection Function:

To test Integrated String for proper calibration and to calibrate analog and digital computer inputs.

II. <u>Bases for Requesting Relief</u>

(A) Date procedure last performed:

July 19, 1981

(B) Date Due:

May 5, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 45 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. It was found that the acceptance criteria for the Loop B reactor coolant flow instrumentation was exceeded. However, the acceptance criteria for the total loop flow was met. The reactor protective system uses the total flow in its calculations.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 10.

(B) Structure/System/Component/Parameter Effected:

Reactor Protection System, reactor coolant flow instrumentation and associated circuitry.

(C) Test/Calibration/Inspection Procedure Name:

Reactor Protection System Channel D Reactor Coolant Flow Instrument Calibration.

(D) Test/Calibration/Inspection Procedure Number:

IP/1/A/305/1L

(E) Test/Calibration/Inspection Function:

To test Integrated String for proper calibration and to calibrate analog and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 20, 1981

(B) Date Due:

May 6, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 45 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. Both Loop A and Loop B had one of its reactor coolant flow instrumentation not meet its acceptance criteria. However, the acceptance criteria for total flow was met. The reactor protective system uses the total flow in its calculations.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 21.

(B) Structure/System/Component/Parameter Effected:

Engineered Safeguards Protection System, Reactor Building pressure switch and contact buffer.

(C) Test/Calibration/Inspection Procedure Name:

Engineered Safeguards System Analog Channel A Reactor Building pressure switch calibration and pressure switch contact buffer test.

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/310/3D

(E) Test/Calibration/Inspection Function:

To calibrate Reactor Building pressure switch instrumentation and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 22, 1981

(B) Date Due:

May 8, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 45 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 21.

(B) Structure/System/Component/Parameter Effected:

Engineered Safeguards Protection System, Reactor Building pressure switch and contact buffer.

(C) Test/Calibration/Inspection Procedure Name:

Engineered Safeguards System Analog Channel B Reactor Building pressure switch calibration and pressure switch contact buffer test.

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/310/4D

(E) Test/Calibration/Inspection Function:

To calibrate Reactor Building pressure switch instrumentation and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 22, 1981

(B) Date Due:

May 8, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 45 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 21.

(B) Structure/System/Component/Parameter Effected:

Engineered Safeguards Protection System, Reactor Building pressure switch and contact buffer.

(C) Test/Calibration/Inspection Procedure Name:

Engineered Safeguards System Analog Channel C Reactor Building pressure switch calibration and pressure switch contact buffer test.

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/310/5D

(E) Test/Calibration/Inspection Function:

To calibrate Reactor Building pressure switch instrumentation and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 23, 1981

(B) Date Due:

May 9, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 45 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Items 15b, 17b, 19.

(B) Structure/System/Component/Parameter Effected:

Engineered Safeguards Protection System, narrow range Reactor Building Pressure Instrumentation and associated circuitry.

(C) Test/Calibration/Inspection Procedure Name:

Engineered Safeguards System Reactor Building narrow range pressure calibration and Analog Channel A test.

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/310/3C

(E) Test/Calibration/Inspection Function:

To calibrate Reactor Building pressure instrument components, analog and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 22, 1981

(B) Date Due:

May 8, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 45 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Items 15b, 17b, 19.

(B) Structure/System/Component/Parameter Effected:

Engineered Safeguards Protection System, narrow range Reactor Building Pressure Instrumentation and associated circuitry.

(C) Test/Calibration/Inspection Procedure Name:

Engineered Safeguards System Reactor Building narrow range pressure calibration and Analog Channel B test.

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/310/4C

(E) Test/Calibration/Inspection Function:

To calibrate Reactor Building pressure instrument components, analog and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 22, 1981

(B) Date Due:

May 8, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 45 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Items 15b, 17b, 19.

(B) Structure/System/Component/Parameter Effected:

Engineered Safeguards Protection System, narrow range Reactor Building Pressure Instrumentation and associated circuitry.

(C) Test/Calibration/Inspection Procedure Name:

Engineered Safeguards System Reactor Building narrow range pressure calibration and Analog Channel C test.

(D) Test/Calibration/Inspection Procedure Number:

IP/0/A/310/5C

(E) Test/Calibration/Inspection Function:

To calibrate Reactor Building pressure instrument components, analog and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 23, 1981

(B) Date Due:

May 9, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 45 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Items 15a, 17a.

(B) Structure/System/Component/Parameter Effected:

Engineered Safeguards Protection System, Reactor Coolant Pressure Instrumentation and associated circuitry.

(C) Test/Calibration/Inspection Procedure Name:

Engineered Safeguards System Analog Channel A Reactor Coolant pressure channel calibration.

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/310/3B

(E) Test/Calibration/Inspection Function:

To calibrate the reactor coolant pressure channel instrument components, analog and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 31, 1981

(B) Date Due:

May 17, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 30 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Items 15a, 17a.

(B) Structure/System/Component/Parameter Effected:

Engineered Safeguards Protection System, Reactor Coolant Pressure Instrumentation and associated circuitry.

(C) Test/Calibration/Inspection Procedure Name:

Engineered Safeguards System Analog Channel B Reactor Coolant Pressure Channel Calibration.

(D) Test/Calibration/Inspection Procedure Number:

IP/0/A/310/4B

(E) Test/Calibration/Inspection Function:

To calibrate the reactor coolant pressure channel instrument components, analog and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 31, 1981

(B) Date Due:

May 17, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 30 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Items 15a, 17a.

(B) Structure/System/Component/Parameter Effected:

Engineered Safeguards Protection System, Reactor Coolant Pressure Instrumentation and associated circuitry.

(C) Test/Calibration/Inspection Procedure Name:

Engineered Safeguards System Analog Channel C Reactor Coolant Pressure Channel Calibration.

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/310/5B

(E) Test/Calibration/Inspection Function:

To calibrate the reactor coolant pressure channel instrument components, analog and digital computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

July 31, 1981

(B) Date Due:

May 17, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 30 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specifications Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 26.

(B) Structure/System/Component/Parameter Effected:

Water Level Indicating System within the pressurizer.

(C) Test/Calibration/Inspection Procedure Name:

Reactor Coolant Pressurizer Level

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/200/10

(E) Test/Calibration/Inspection Function:

To calibrate all instrument components associated with the Pressurizer Water Level Indicating System, and the Analog Computer points.

II. Bases for Requesting Relief

(A) Date procedure last performed:

August 10, 1981

(B) Date Due:

May 27, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 25 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 39.

(B) Structure/System/Component/Parameter Effected:

Water Level Indicating System within the Steam Generators.

(C) Test/Calibration/Inspection Procedure Name:

Feedwater System Steam Generator full range and startup level instrumentation calibration.

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/275/5P

(E) Test/Calibration/Inspection Function:

To calibrate all instrument components associated with the Steam Generator Water Level Indicating System, and the Analog Computer inputs.

II. Bases for Requesting Relief

(A) Date procedure last performed:

August 11, 1981

(B) Date Due:

May 28, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 25 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 39.

(B) Structure/System/Component/Parameter Effected:

Steam Generator Operating Level Instrumentation System.

(C) Test/Calibration/Inspection Procedure Name:

Steam Generator Temperature Compensated Operating Level Instrument Calibration.

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/275/5Q

(E) Test/Calibration/Inspection Function:

To calibrate all instrument components associated with the Steam Generator Operating Level System.

II. Bases for Requesting Relief

(A) Date procedure last performed:

August 12, 1981

(B) Date Due:

May 29, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 25 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 23.

(B) Structure/System/Component/Parameter Effected:

Control Rod Drive Position Indicator

(C) Test/Calibration/Inspection Procedure Name:

Control Rod Drive Absolute Position Indication Calibration Absolute Position Indication and Relative Position Group Average Calibration - Asymmetric Rod Calibration

(D) Test/Calibration/Inspection Procedure Number:

IP/0/A/340/4

(E) Test/Calibration/Inspection Function:

To calibrate Control Rod Drive Position Indicating System, and calibrate rod misalignment channel.

II. Bases for Requesting Relief

(A) Date procedure last performed:

August 25, 1981

(B) Date Due:

June 11, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 15 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 24.

(B) Structure/System/Component/Parameter Effected:

Control Rod Drive Position Indication

(C) Test/Calibration/Inspection Procedure Name:

Control Rod Drive Relative Position Indicator Amplifier, Group Average Meters and Position Indicator Meters.

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/340/5

(E) Test/Calibration/Inspection Function:

To calibrate Control Rod Drive Position Indicating System.

II. Bases for Requesting Relief

(A) Date procedure last performed:

August 26, 1981

(B) Date Due:

June 12, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 15 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 38.

(B) Structure/System/Component/Parameter Effected:

Emergency Sump Level Indication System

(C) Test/Calibration/Inspection Procedure Name:

Low Pressure Injection System Reactor Building Emergency Sump Level Instrument Calibration

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/203/IE

(E) Test/Calibration/Inspection Function:

To calibrate the Reactor Building Emergency Sump Level Instrumentation System.

II. Bases for Requesting Relief

(A) Date procedure last performed:

August 30, 1981

(B) Date Due:

June 16, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 15 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This instrument channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length.

In addition, the results of the previous calibration was investigated. The acceptance criterial was not met. However, the drift was in the conservative direction.

This instrument is used to indicate amount of water that has leaked out of the reactor coolant system and into the reactor building. There are redundant systems to indicate leakage from the reactor coolant system. These are: the reactor building air particular monitor, iodine monitors, gaseous monitoring area monitors and water inventory balances.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specifications Effected:

Section 4.5.1.2.2 - Power operated valve, component tests of the Emergency Core Cooling System.

(B) Structure/System/Component/Parameter Effected:

Electrically operated valves (LP-9, 10, 12, 14, 17, 18) of the Low Pressure Injection and Core Flooding System.

(C) Test/Calibration/Inspection Procedure Name:

Low Pressure Injection System Power Operated Valves Manual Operability Test.

(D) Test/Calibration/Inspection Procedure Number:

PT/0/A/203/5A

(E) Test/Calibration/Inspection Function:

To verify manual operability of certain power operated valves on the Low Pressure Injection System.

II. Bases for Requesting Relief

(A) Date procedure last performed:

August 20, 1981

(B) Date Due:

June 6, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 20 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

The purpose of this procedure is to verify that valves LP-9, 10, 12, 14, 17, 18 can be manually operated. These valves are all electrically operated valves and only valves LP-17 and 18 are part of the Engineered Safeguards System. LP-17, 18 were successfully stroke tested on October 24, 1982, and on March 3, 1983 valves LP-12, 14 were successfully stroke tested. Valves LP-9, 10 were successfully stroke tested on March 4, 1983. It is highly unlikely that these valves will fail to operate electrically when required however, if there is a forced outage of sufficient duration, every effort will be made to perform this test at that time.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specifications Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 25b.

(B) Structure/System/Component/Parameter Effected:

Core flood tank level instrumentation.

(C) Test/Calibration/Inspection Procedure Name:

Core Flood Tank Level Instrument Calibration

(D) Test/Calibration/Inspection Procedure Number:

IP/0/A/201/1A

(E) Test/Calibration/Inspection Function:

To calibrate the core flood tank level instrumentation system.

II. Bases for Requesting Relief

(A) Date procedure last performed:

April 28, 1981

(B) Date Due:

July 13, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have been shutdown for refueling by the due date, if there are no forced outages until refueling.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.

III. Justification for Relief

This channel is subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift is expected to remain within acceptable limits until recalibration can be performed during the upcoming refueling on the next forced outage of sufficient length. Substantial calibraiton shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

In addition, the results of the previous calibration was investigated. The acceptance criteria was met.