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10 CFR 50.90

August 21, 2006

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Limerick Generating Station, Units 1 and 2
Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Subject: License Amendment Request - Proposed Technical Specifications Change to Relocate Surveillance Test Intervals to a Licensee-Controlled Program (Risk-Informed Initiative 5b); Transmittal of Revised Technical Specifications and Bases Marked-up Pages

Reference: (1) Letter from M. P. Gallagher, Exelon Generation Company, LLC, to U. S. Nuclear Regulatory Commission, dated June 11, 2004.

In Reference 1, Exelon Generation Company, LLC (Exelon), requested a change to the Technical Specifications (TS), Appendix A, of Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2, respectively. The proposed change relocates the surveillance test intervals (STIs) of various TS surveillance requirements from the TS to a new licensee program, the Surveillance Frequency Control Program, which is being added to the Administrative Controls section of TS. This license amendment request (LAR) was submitted as a pilot in support of the Boiling Water Reactor Owners' Group (BWROG) Risk-Informed Initiative 5b, "Relocate Surveillance Test Intervals to Licensee Control."

TS/Bases marked-up pages originally submitted in Reference 1 have been changed for clarification and consistency to ensure that only surveillance frequencies are deleted from TS and relocated to the licensee-controlled program. Attachment 1 provides the revised TS/Bases marked-up pages for LGS, Unit 1. Attachment 2 provides the revised TS/Bases marked-up pages for LGS, Unit 2. Only the specific TS/Bases marked-up pages which have changed from the TS/Bases marked-up pages provided in Reference 1 are being submitted in the attachments.

Exelon has concluded that the information provided in this letter does not impact the conclusions of the: (1) Technical Analysis, (2) No Significant Hazards Consideration under the standards set forth in 10 CFR 50.92(c), or (3) Environmental Consideration as provided in Reference 1.

There are no regulatory commitments contained within this letter.

If you have any questions or require additional information, please contact Glenn Stewart at 610-765-5529.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 21th day of August, 2006.

Respectfully,

Pamela B. Cowan
Director - Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
Attachment 1: LGS Unit 1 Revised Technical Specifications and Bases Marked-up Pages
Attachment 2: LGS Unit 2 Revised Technical Specifications and Bases Marked-up Pages

cc:	Regional Administrator - NRC Region I	w/attachments
	NRC Senior Resident Inspector - Limerick Generating Station	"
	NRC Project Manager, NRR - Limerick Generating Station	"
	NRC Project Manager - BWROG	"
	NRC Project Manager - RITS Task Force	"
	Director, Bureau of Radiation Protection - Pennsylvania Department of Environmental Protection	"

ATTACHMENT 1

LICENSE AMENDMENT REQUEST

LIMERICK GENERATING STATION, UNITS 1 AND 2
DOCKET NOS. 50-352 AND 50-353

RELOCATION OF SURVEILLANCE TEST INTERVALS FROM THE TECHNICAL SPECIFICATIONS TO A LICENSEE-CONTROLLED PROGRAM

REVISED UNIT 1 TECHNICAL SPECIFICATIONS AND BASES MARKED-UP PAGES FOR THE PROPOSED CHANGE

3/4 3-52
3/4 3-61
3/4 3-88
3/4 4-17
3/4 4-19
3/4 5-9
3/4 6-18
3/4 7-5
3/4 9-4
3/4 9-7
B 3/4 3-1
B 3/4 3-2
B 3/4 3-3
B 3/4 3-4
B 3/4 3-5
B 3/4 3-6
B 3/4 4-3d
6-14d

INSTRUMENTATION

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION;

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each **of the required** RCIC system actuation instrumentation channel**s** shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies ~~shown in Table 4.3.5.1-1~~ **specified in the Surveillance Frequency Control Program. CHANNEL CHECK and CHANNEL CALIBRATION are not required for manual initiation.**

4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed ~~at least once per 24 months~~ **in accordance with the Surveillance Frequency Control Program**.

TABLE 4.3.6-1
CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK(h)	CHANNEL FUNCTIONAL TEST(h)	CHANNEL CALIBRATION(a)(h)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	N.A.	~~Q~~(c)	~~R~~	1*
b. Inoperative	N.A.	~~Q~~(c)	N.A.	1*
c. Downscale	N.A.	~~Q~~(c)	~~R~~	1*
2. APRM				
a. Simulated Thermal Power-Upscale	N.A.	~~SA~~	~~R~~	1
b. Inoperative	N.A.	~~SA~~	N.A.	1, 2
c. Neutron Flux - Downscale	N.A.	~~SA~~	~~R~~	1
d. Simulated Thermal Power - Upscale (Setdown)	N.A.	~~SA~~	~~R~~	2
e. Recirculation Flow - Upscale	N.A.	~~SA~~	~~R~~	1
f. LPRM Low Count	N.A.	~~SA~~	~~R~~	1, 2
3. SOURCE RANGE MONITORS				
a. Detector not full in	N.A.	~~M(d)~~(e)~~, W(f)~~	N.A.	2, 5
b. Upscale	N.A.	~~M(d)~~(e)~~, W(f)~~	~~R~~	2, 5
c. Inoperative	N.A.	~~M(d)~~(e)~~, W(f)~~	N.A.	2, 5
d. Downscale	N.A.	~~M(d)~~(e)~~, W(f)~~	~~R~~	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	N.A.	~~W~~	N.A.	2, 5
b. Upscale	N.A.	~~W~~	~~R~~	2, 5
c. Inoperative	N.A.	~~W~~	N.A.	2, 5
d. Downscale	N.A.	~~W~~	~~R~~	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level - High	N.A.	~~Q~~	~~R~~	1, 2, 5**
6. DELETED	~~DELETED~~	~~DELETED~~	~~DELETED~~	~~DELETED~~
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	~~R~~(g)	N.A.	3, 4

INSTRUMENTATION

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

a. In OPERATIONAL CONDITION 2*, three.

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3, and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least three source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

1. CHANNEL CHECK ~~at least once per:~~ **in accordance with the Surveillance Frequency Control Program:**

a) ~~12 hours~~ in CONDITION 2*, AND

b) ~~24 hours~~ in CONDITION 3 or 4.

2. CHANNEL CALIBRATION** ~~at least once per 24 months~~ **in accordance with the Surveillance Frequency Control Program**.

b. Performance of a CHANNEL FUNCTIONAL TEST ~~at least once per 31 days~~ **in accordance with the Surveillance Frequency Control Program**.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 3.0 cps*** with the detector fully inserted.

*With IRM's on range 2 or below.
**Neutron detectors may be excluded from CHANNEL CALIBRATION.
***May be reduced, provided the source range monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1.

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS IS REQUIRED
1. Gross Beta and Gamma Activity Determination	~~At least once per 72 hours~~ **In accordance with the Surveillance Frequency Control Program.**	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	~~At least once per 31 days~~ **In accordance with the Surveillance Frequency Control Program.**	1
3. Radiochemical for $\bar{E}$ Determination	~~At least once per 6 months*~~ **In accordance with the Surveillance Frequency Control Program.***	1
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1**, 2**, 3**, 4**
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135, and Kr-88	~~At least once per 31 days~~ **In accordance with the Surveillance Frequency Control Program.**	1

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

** Until the specific activity of the primary coolant system is restored to within its limits.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and ~~at least once per 30 minutes~~ **in accordance with the Surveillance Frequency Control Program** during system heatup.

4.4.6.1.3 DELETED

4.4.6.1.4 DELETED

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 80°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 100°F, ~~at least once per 12 hours~~ **in accordance with the Surveillance Frequency Control Program**.

   2. ≤ 90°F, ~~at least once per 30 minutes~~ **in accordance with the Surveillance Frequency Control Program**.

b. Within 30 minutes prior to and ~~at least once per 30 minutes~~ **in accordance with the Surveillance Frequency Control Program** during tensioning of the reactor vessel head bolting studs.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying the water level to be greater than or equal to, as applicable:

a. 22'0" ~~at least once per 24 hours~~ **in accordance with the Surveillance Frequency Control Program**.

b. 16'0" ~~at least once per 12 hours~~ **in accordance with the Surveillance Frequency Control Program**.

4.5.3.2 With the suppression chamber level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, ~~at least once per 12 hours~~ **in accordance with the Surveillance Frequency Control Program**:

a. Verify the required conditions of Specification 3.5.3b. to be satisfied, or

b. Verify footnote conditions * to be satisfied.

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shall be demonstrated OPERABLE ~~at least once per 24 months~~ **in accordance with the Surveillance Frequency Control Program** by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 A representative sample of reactor instrumentation line excess flow check valves shall be demonstrated OPERABLE ~~at least once per 24 months~~ **in accordance with the Surveillance Frequency Control Program**, such that each valve is tested ~~at least once every 120 months~~ **in accordance with the Surveillance Frequency Control Program**, by verifying that the valve checks flow.*

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. ~~At least once per 31 days~~ **In accordance with the Surveillance Frequency Control Program** by verifying the continuity of the explosive charge.

b. ~~At least once per 24 months~~ **In accordance with the Surveillance Frequency Control Program** by removing the explosive squib from the explosive valve, such that each explosive squib in each explosive valve will be tested ~~at least once every 120 months~~ **in accordance with the Surveillance Frequency Control Program**, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and/or operating life, as applicable.

*The reactor vessel head seal leak detection line (penetration 29A) excess flow check valve is not required to be tested pursuant to this requirement.

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The spray pond shall be OPERABLE with:

a. A minimum pond water level at or above elevation 250' 10" Mean Sea Level, and

b. A pond water temperature of less than or equal to 88°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and *.

ACTION:

With the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 4 or 5, declare the RHRSW system and the emergency service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

c. In OPERATIONAL CONDITION *, declare the emergency service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The spray pond shall be determined OPERABLE:

a. By verifying the pond water level to be greater than its limit ~~at least once per 24 hours~~ **in accordance with the Surveillance Frequency Control Program**.

b. By verifying the water surface temperature (within the upper two feet of the surface) to be less than or equal to 88°F:

1. ~~at least once per 4 hours~~ **in accordance with the Surveillance Frequency Control Program** when the spray pond temperature is greater than or equal to 80°F; and

2. ~~at least once per 2 hours~~ **in accordance with the Surveillance Frequency Control Program** when the spray pond temperature is greater than or equal to 85°F; and

3. ~~at least once per 24 hours~~ **in accordance with the Surveillance Frequency Control Program** when the spray pond temperature is greater than 32°F.

c. By verifying all piping above the frost line is drained:

1. within one (1) hour after being used when ambient air temperature is below 40°F; or

2. when ambient air temperature falls below 40°F if the piping has not been previously drained.

*When handling irradiated fuel in the secondary containment.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST ~~at least once per 7 days~~ **in accordance with the Surveillance Frequency Control Program**.

c. Verifying that the channel count rate is at least 3.0 cps:*

1. Prior to control rod withdrawal,

2. Prior to and ~~at least once per 12 hours~~ **in accordance with the Surveillance Frequency Control Program** during CORE ALTERATIONS, and

3. ~~At least once per 24 hours~~ **In accordance with the Surveillance Frequency Control Program**.

d. Verifying, within 8 hours prior to and ~~at least once per 12 hours~~ **in accordance with the Surveillance Frequency Control Program** ~~during~~, that the RPS circuitry "shorting links" have been removed during:

1. The time any control rod is withdrawn**, unless adequate shutdown margin has been demonstrated, or

2. Shutdown margin demonstrations.

*May be reduced, provided the source range monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1. These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core.

**Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling floor personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

When direct communication between the control room and refueling floor personnel cannot be maintained, immediately suspend CORE ALTERATIONS.*

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated ~~at least once per 12 hours~~ **in accordance with the Surveillance Frequency Control Program** during CORE ALTERATIONS.*

*Except movement of control rods with their normal drive system.

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The APRM system is divided into four APRM chanels and four 2-Out-Of-4 Voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed.

The system meets the intent of IEEE-279 for nuclear power plant protection systems. ~~Specified~~ ~~s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** ~~and surveillance~~ and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System" and NEDC-32410P-A, “Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMARC PRNM) Retrofit Plus Option III Stability Trip Function.” The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

Actions a, b and c define the action(s) required when RPS channels are discovered to be inoperable. For those actions, separate entry condition is allowed for each inoperable RPS channel. Separate entry means that the allowable time clock(s) for actions a, b or c start upon discovery of inoperability for that specific channel. Restoration of an inoperable RPS channel satisfies only the action statements for that particular channel. Action statement(s) for remaining inoperable channel(s) must be met according to their original entry time.

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (NEDC-30851P-A and NEDC-32410P-A) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided that the associated Function’s (identified as a “Functional Unit” in Table 3.3.1-1) inoperable channel is in one trip system and the Function still maintains RPS trip capability.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance.

~~Specified~~ ~~s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage times have been determined in accordance with NEDC-30851P, Supplement 2, "Technical Specification Improvement Analysis for BWR Instrumentation Common to RPS and ECCS Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (SER) (letter to D.N. Grace from C.E. Rossi dated January 6, 1989) and NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," as approved by the NRC and documented in the NRC SER (letter to S.D. Floyd from C.E. Rossi dated June 18, 1990).

Automatic closure of the MSIVs upon receipt of a high-high radiation signal from the Main Steam Line Radiation Monitoring System was removed as the result of an analysis performed by General Electric in NEDO-31400A. The NRC approved the results of this analysis as documented in the SER (letter to George J. Beck, BWR Owner's Group from A.C. Thadani, NRC, dated May 15, 1991).

Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10-second diesel startup and the 3 second load center loading delay. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions.

Response time testing for sensors are not required based on the analysis in NEDO 32291-A. Response time testing of the remaining channel components is required as noted in Table 3.3.2-3.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses. Primary containment isolation valves that are actuated by the isolation signals specified in Technical Specification Table 3.3.2-1 are identified in Technical Requirements Manual Table 3.6.3-1.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

3/4.3.3 EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION (Continued)

~~Specified~~ ~~s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage times have been determined in accordance with NEDC-30936P, Parts 1 and 2, "Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation)," as approved by the NRC and documented in the SER (letter to D. N. Grace from A. C. Thadani dated December 9, 1988 (Part 1) and letter to D. N. Grace from C. E. Rossi dated December 9, 1988 (Part 2)).

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power for energizing various components such as pump motors, motor operated valves, and the associated control components. If the loss of power instrumentation detects that voltage levels are too low, the buses are disconnected from the offsite power sources and connected to the onsite diesel generator (DG) power sources. The loss of power relays in each channel have sufficient overlapping detection characteristics and functionality to permit operation subject to the conditions in Action Statement 37. Bases 3/4.8.1, 3/4.8.2, and 3/4.8.3 provide discussion regarding parametric bounds for determining operability of the offsite sources. Those Bases assume that the loss of power relays are operable. With an inoperable 127Z-11X0X relay, the grid voltage is monitored to 230kV (for the 101 Safeguard Bus Source) or 525kV (for the 201 Safeguard Bus Source) to increase the margin for the operation of the 127Z-11X0X relay.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a supplement to the reactor trip. During turbine trip and generator load rejection events, the EOC-RPT will reduce the likelihood of reactor vessel level decreasing to level 2. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

~~Specified~~ ~~s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel. This instrumentation does not provide actuation of any of the emergency core cooling equipment.

~~Specified~~ ~~s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage times have been specified in accordance with recommendations made by GE in their letter to the BWR Owner's Group dated August 7, 1989, SUBJECT: "Clarification of Technical Specification changes given in ECCS Actuation Instrumentation Analysis."

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

~~Specified~~ ~~s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage times have been determined in accordance with NEDC-30851P, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," as approved by the NRC and documented in the SER (letter to D. N. Grace from C. E. Rossi dated September 22, 1988).

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

The ~~specified~~ surveillance interval for the Main Control Room Normal Fresh Air Supply Radiation Monitor ~~has been~~ **is** determined in accordance with **the Surveillance Frequency Control Program** ~~GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).~~

3/4.3.7.2 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE UFSAR.

3/4.3.7.3 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4.3.7.4 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown system instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50, Appendix A.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

Drywell and containment hydrogen and oxygen analyzers are Category I instruments provided to detect high hydrogen or oxygen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions.

When two hydrogen monitor channels are inoperable, one hydrogen monitor channel must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time is based on the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit; the length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding this limit; and the availability of the hydrogen recombiners, the Containment Purge System, and the Post Accident Sampling Systems.

3/4.3.7.7 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE TRM.

3/4.3.7.8 CHLORINE AND TOXIC GAS DETECTION SYSTEMS

The OPERABILITY of the chlorine and toxic gas detection systems ensures that an accidental chlorine and/or toxic gas release will be detected promptly and the necessary protective actions will be automatically initiated for chlo-rine and manually initiated for toxic gas to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the chlorine isolation mode of operation to provide the required protection. Upon detection of a high concentration of toxic gas, the control room emergency ventilation system will manually be placed in the chlorine isolation mode of operation to provide the required protection. The detection systems required by this speci-fication are consistent with the recommendations of Regulatory Guide 1.95, "Pro-tection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release," February 1975.

There are three toxic gas detection subsystems. The high toxic chemical concentration alarm in the Main Control Room annunciates when two of the three subsystems detect a high toxic gas concentration. An Operate/Inop keylock switch is provided for each subsystem which allows an individual subsystem to be placed in the tripped condition. Placing the keylock switch in the INOP position initiates one of the two inputs required to initiate the alarm in the Main Control Room.

~~Specified~~ ~~s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

3/4.3.7.9 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE TRM.

SURVEILLANCE REQUIREMENTS (Continued)

SR 4.4.3.1.b
This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm setpoint and relative accuracy of the instrument string. ~~The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.~~

SR 4.4.3.1.c
The SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. ~~The Frequency of 24 months is for a typical refueling cycle and considers channel reliability. Operating experience has proven this frequency to be acceptable.~~

SR 4.4.3.1.d
This SR provides a routine check of primary containment pressure and temperature for indirect evidence of RCS leakage. ~~Operating experience has proven this frequency to be acceptable.~~

REFERENCES

1. LGS UFSAR, Section 5.2.5.1.
2. Regulatory Guide 1.45, May 1973.
3. LGS UFSAR, Section 5.2.5.2.1.3.
4. LGS UFSAR, Section 5.2.5.2.1.5.
5. LGS UFSAR, Section 5.2.5.2.1.4.
6. LGS UFSAR, Section 5.2.5.2.1.1(2).
7. GEAP-5620, April 1968
8. NUREG-75/067, October 1975.
9. LGS UFSAR, Section 5.2.5.6.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. The limit of 2 gpm increase in UNIDENTIFIED LEAKAGE over a 24-hour period and the monitoring of drywell floor drain sump and drywell equipment drain tank flow rate at least once every eight (8) hours conforms with NRC staff positions specified in NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," as revised by NRC Safety Evaluation dated March 6, 1990. The ACTION requirement for the 2 gpm increase in UNIDENTIFIED LEAKAGE limit ensures that such leakage is identified or a plant shutdown is initiated to allow further investigation and corrective action. Once identified, reactor operation may continue dependent upon the impact on total leakage.

i. Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries For Stationary Applications,” of the following:

a. Actions to restore battery cells with float voltage < 2.13 volts, and

b. Actions to equalize and test battery cells that have been discovered with electrolyte level below the minimum established design limit.

j. Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 0.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

ATTACHMENT 2

LICENSE AMENDMENT REQUEST

LIMERICK GENERATING STATION, UNITS 1 AND 2
DOCKET NOS. 50-352 AND 50-353

RELOCATION OF SURVEILLANCE TEST INTERVALS FROM THE TECHNICAL SPECIFICATIONS TO A LICENSEE-CONTROLLED PROGRAM

REVISED UNIT 2 TECHNICAL SPECIFICATIONS AND BASES MARKED-UP PAGES FOR THE PROPOSED CHANGE

3/4 3-52
3/4 3-61
3/4 3-88
3/4 4-19
3/4 5-9
3/4 6-18
3/4 7-5
3/4 9-4
3/4 9-7
B 3/4 3-1
B 3/4 3-2
B 3/4 3-3
B 3/4 3-4
B 3/4 3-5
B 3/4 3-6
B 3/4 4-3d
6-14d

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each **of the required** RCIC system actuation instrumentation channel**s** shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies ~~shown in Table 4.3.5.1-1~~ **specified in the Surveillance Frequency Control Program. CHANNEL CHECK and CHANNEL CALIBRATION are not required for manual initiation**.

4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed ~~at least once per 24 months~~ **in accordance with the Surveillance Frequency Control Program**.

<u>TABLE 4.3.6-1</u>
<u>CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS</u>

TRIP FUNCTION	CHANNEL CHECK **(h)**	CHANNEL FUNCTIONAL TEST **(h)**	CHANNEL CALIBRATION(a) **(h)**	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. <u>ROD BLOCK MONITOR</u>				
a. Upscale	N.A.	~~Q~~(c)	~~R~~	1*
b. Inoperative	N.A.	~~Q~~(c)	N.A.	1*
c. Downscale	N.A.	~~Q~~(c)	~~R~~	1*
2. <u>APRM</u>				
a. Simulated Thermal Power - Upscale	N.A.	~~SA~~	~~R~~	1
b. Inoperative	N.A.	~~SA~~	N.A.	1, 2
c. Neutron Flux - Downscale	N.A.	~~SA~~	~~R~~	1
d. Simulated Thermal Power - Upscale (Setdown)	N.A.	~~SA~~	~~R~~	2
e. Recirculation Flow - Upscale	N.A.	~~SA~~	~~R~~	1
f. LPRM Low Count	N.A.	~~SA~~	~~R~~	1, 2
3. <u>SOURCE RANGE MONITORS</u>				
a. Detector not full in	N.A.	~~M(d)~~(e)~~, W(f)~~	N.A.	2, 5
b. Upscale	N.A.	~~M(d)~~(e)~~, W(f)~~	~~R~~	2, 5
c. Inoperative	N.A.	~~M(d)~~(e)~~, W(f)~~	N.A.	2, 5
d. Downscale	N.A.	~~M(d)~~(e)~~, W(f)~~	~~R~~	2, 5
4. <u>INTERMEDIATE RANGE MONITORS</u>				
a. Detector not full in	N.A.	~~W~~	N.A.	2, 5
b. Upscale	N.A.	~~W~~	~~R~~	2, 5
c. Inoperative	N.A.	~~W~~	N.A.	2, 5
d. Downscale	N.A.	~~W~~	~~R~~	2, 5
5. <u>SCRAM DISCHARGE VOLUME</u>				
a. Water Level - High	N.A.	~~Q~~	~~R~~	1, 2, 5**
6. DELETED	~~DELETED~~	~~DELETED~~	~~DELETED~~	~~DELETED~~
7. <u>REACTOR MODE SWITCH SHUTDOWN POSITION</u>	N.A.	~~R~~(g)	N.A.	3, 4

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

a. In OPERATIONAL CONDITION 2*, three.

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*#, 3, and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least three source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

1. CHANNEL CHECK ~~at least once per:~~ **in accordance with the Surveillance Frequency Control Program:**

a) ~~12 hours~~ in CONDITION 2*, and
b) ~~24 hours~~ in CONDITION 3 or 4.

2. CHANNEL CALIBRATION** ~~at least once per 24 months~~ **in accordance with the Surveillance Frequency Control Program**.

b. Performance of a CHANNEL FUNCTIONAL TEST ~~at least once per 31 days~~ **in accordance with the Surveillance Frequency Control Program**.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 3.0 cps*** with the detector fully inserted.#

*With IRM's on range 2 or below.
**Neutron detectors may be excluded from CHANNEL CALIBRATION.
***May be reduced, provided the source range monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1.
#During initial startup test program, SRM detectors may be partially withdrawn prior to IRM on-scale indication provided that the SRM channels remain on scale above 100 cps and respond to changes in the neutron flux.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and ~~at least once per 30 minutes~~ **in accordance with the Surveillance Frequency Control Program** during system heatup.

4.4.6.1.3 DELETED

4.4.6.1.4 DELETED

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 100°F, ~~at least once per 12 hours~~ **in accordance with the Surveillance Frequency Control Program**.

   2. ≤ 90°F, ~~at least once per 30 minutes~~ **in accordance with the Surveillance Frequency Control Program**.

b. Within 30 minutes prior to and ~~at least once per 30 minutes~~ **in accordance with the Surveillance Frequency Control Program** during tensioning of the reactor vessel head bolting studs.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying the water level to be greater than or equal to, as applicable:

a. 22'0" ~~at least once per 24 hours~~ **in accordance with the Surveillance Frequency Control Program**.

b. 16'0" ~~at least once per 12 hours~~ **in accordance with the Surveillance Frequency Control Program**.

4.5.3.2 With the suppression chamber level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, ~~at least once per 12 hours~~ **in accordance with the Surveillance Frequency Control Program**:

a. Verify the required conditions of Specification 3.5.3b. to be satisfied, or

b. Verify footnote conditions * to be satisfied.

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shall be demonstrated OPERABLE ~~at least once per 24 months~~ **in accordance with the Surveillance Frequency Control Program** by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 A representative sample of instrumentation line excess flow check valves shall
be demonstrated OPERABLE ~~at least once per 24 months~~ **in accordance with the Surveillance Frequency Control Program**, such that each valve is tested ~~at least once per every 120 months~~ **in accordance with the Surveillance Frequency Control Program**, by verifying that the valve checks flow.*

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. ~~At least once per 31~~ **In accordance with the Surveillance Frequency Control Program** days by verifying the continuity of the explosive charge.

b. ~~At least once per 24 months~~ **In accordance with the Surveillance Frequency Control Program** by removing the explosive squib from the explosive valve, such that each explosive squib in each explosive valve will be tested ~~at least once per 120 months~~ **in accordance with the Surveillance Frequency Control Program**, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and/or operating life, as applicable.

*The reactor vessel head seal leakage detection line (penetration 29A) excess flow check valve is not required to be tested pursuant to this requirement.

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The spray pond shall be OPERABLE with:

a. A minimum pond water level at or above elevation 250'-10" Mean Sea Level, and

b. A pond water temperature of less than or equal to 88°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and *.

ACTION:

With the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 4 or 5, declare the RHRSW system and the emergency service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

c. In OPERATIONAL CONDITION *, declare the emergency service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The spray pond shall be determined OPERABLE:

a. By verifying the pond water level to be greater than its limit ~~at least once per 24 hours~~ **in accordance with the Surveillance Frequency Control Program**.

b. By verifying the water surface temperature (within the upper two feet of the surface) to be less than or equal to 88°F:

   1. ~~at least once per 4 hours~~ **in accordance with the Surveillance Frequency Control Program** when the spray pond temperature is greater than or equal to 80°F; and

   2. ~~at least once per 2 hours~~ **in accordance with the Surveillance Frequency Control Program** when the spray pond temperature is greater than or equal to 85°F; and

   3. ~~at least once per 24 hours~~ **in accordance with the Surveillance Frequency Control Program** when the spray pond temperature is greater than 32°F.

c. By verifying all piping above the frost line is drained:

   1. within one (1) hour after being used when ambient air temperature is below 40°F; or

   2. when ambient air temperature falls below 40°F if the piping has not been previously drained.

*When handling irradiated fuel in the secondary containment.

b. Performance of a CHANNEL FUNCTIONAL TEST ~~at least once per 7 days~~ **in accordance with the Surveillance Frequency Control Program**.

c. Verifying that the channel count rate is at least 3.0 cps:*

1. Prior to control rod withdrawal,

2. Prior to and ~~at least once per 12 hours~~ **in accordance with the Surveillance Frequency Control Program** during CORE ALTERATIONS, and

3. ~~At least once per 24 hours~~ **In accordance with the Surveillance Frequency Control Program**.

d. Verifying, within 8 hours prior to and ~~at least once per 12 hours during~~ **in accordance with the Surveillance Frequency Control Program**, that the RPS circuitry "shorting links" have been removed during:

1. The time any control rod is withdrawn**, unless adequate shutdown margin has been demonstrated, or

2. Shutdown margin demonstrations.

*May be reduced, provided the source range monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1. These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core.

**Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling floor personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

When direct communication between the control room and refueling floor personnel cannot be maintained, immediately suspend CORE ALTERATIONS.*

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated ~~at least once per 12 hours~~ **in accordance with the Surveillance Frequency Control Program** during CORE ALTERATIONS.*

*Except movement of control rods with their normal drive system.

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The APRM system is divided into four APRM channels and four 2-Out-Of-4 Voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed.

The system meets the intent of IEEE-279 for nuclear power plant protection systems. ~~Specified s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** ~~and surveillance~~ and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System" and NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function." The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

Actions a, b and c define the action(s) required when RPS channels are discovered to be inoperable. For those actions, separate entry condition is allowed for each inoperable RPS channel. Separate entry means that the allowable time clock(s) for Actions a, b or c start upon discovery of inoperability for that specific channel. Restoration of an inoperable RPS channel satisfies only the action statements for that particular channel. Action statement(s) for remaining inoperable channel(s) must be met according to their original entry time.

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (NEDC-30851P-A and NEDC-32410P-A) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided that the associated Function's (identified as a "Functional Unit" in Table 3.3.1-1) inoperable channel is in one trip system and the Function still maintains RPS trip capability.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance.

~~Specified s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage times have been determined in accordance with NEDC-30851P, Supplement 2, "Technical Specification Improvement Analysis for BWR Instrumentation Common to RPS and ECCS Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (SER) (letter to D.N. Grace from C.E. Rossi dated January 6, 1989) and NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," as approved by the NRC and documented in the NRC SER (letter to S.D. Floyd from C.E. Rossi dated June 18, 1990).

Automatic closure of the MSIVs upon receipt of a high-high radiation signal from the Main Steam Line Radiation Monitoring System was removed as the result of an analysis performed by General Electric in NEDO-31400A. The NRC approved the results of this analysis as documented in the SER (letter to George J. Beck, BWR Owner's Group from A.C. Thadani, NRC, dated May 15, 1991).

Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10-second diesel startup and the 3 second load center loading delay. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions.

Response time testing for sensors are not required based on the analysis in NEDO-32291-A. Response time testing of the remaining channel components is required as noted in Table 3.3.2-3.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses. Primary containment isolation valves that are actuated by the isolation signals specified in Technical Specification Table 3.3.2-1 are identified in Technical Requirements Manual Table 3.6.3-1.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION (Continued)

~~Specified s~~Surveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage times have been determined in accordance with NEDC-30936P, Parts 1 and 2, "Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation)," as approved by the NRC and documented in the SER (letter to D. N. Grace from A. C. Thadani dated December 9, 1988 (Part 1) and letter to D. N. Grace from C. E. Rossi dated December 9, 1988 (Part 2)).

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power for energizing various components such as pump motors, motor operated valves, and the associated control components. If the loss of power instrumentation detects that voltage levels are too low, the buses are disconnected from the offsite power sources and connected to the onsite diesel generator (DG) power sources. The loss of power relays in each channel have sufficient overlapping detection characteristics and functionality to permit operation subject to the conditions in Action Statement 37. Bases 3/4.8.1, 3/4.8.2, and 3/4.8.3 provide discussion regarding parametric bounds for determining operability of the offsite sources. Those Bases assume that the loss of power relays are operable. With an inoperable 127Z-11X0X relay, the grid voltage is monitored to 230kV (for the 101 Safeguard Bus Source) or 525kV (for the 201 Safeguard Bus Source) to increase the margin for the operation of the 127Z-11X0X relay.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a supplement to the reactor trip. During turbine trip and generator load rejection events, the EOC-RPT will reduce the likelihood of reactor vessel level decreasing to level 2. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

~~Specified s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel. This instrumentation does not provide actuation of any of the emergency core cooling equipment.

~~Specified s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage times have been specified in accordance with recommendations made by GE in their letter to the BWR Owner's Group dated August 7, 1989, SUBJECT: "Clarification of Technical Specification changes given in ECCS Actuation Instrumentation Analysis."

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

~~Specified s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage time have been determined in accordance with NEDC-30851P, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," as approved by the NRC and documented in the SER (letter to D. N. Grace from C. E. Rossi dated September 22, 1988).

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The *levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variable following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

The ~~specified~~ surveillance interval for the Main Control Room Normal Fresh Air Supply Radiation Monitor ~~has been~~ **is** determined in accordance with **the Surveillance Frequency Control Program** ~~GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specification," as approved by the NRC and documented in the SER (letter to R. D. Binz, IV, from C. E. Rossi dated July 21, 1992)~~.

3/4.3.7.2 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE UFSAR.

3/4.3.7.3 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4.3.7.4 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown system instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50, Appendix A. The Unit 1 RHR transfer switches are included only due to their potential impact on the RHRSW system, which is common to both units.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

Drywell and containment hydrogen and oxygen analyzers are Category I instruments provided to detect high hydrogen or oxygen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions.

When two hydrogen monitor channels are inoperable, one hydrogen monitor channel must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time is based on the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit; the length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding this limit; and the availability of the hydrogen recombiners, the Containment Purge System, and the Post Accident Sampling Systems.

3/4.3.7.7 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE TRM.

3/4.3.7.8 CHLORINE AND TOXIC GAS DETECTION SYSTEMS

The OPERABILITY of the chlorine and toxic gas detection systems ensures that an accidental chlorine and/or toxic gas release will be detected promptly and the necessary protective actions will be automatically initiated for chlorine and manually initiated for toxic gas to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the chlorine isolation mode of operation to provide the required protection. Upon detection of a high concentration of toxic gas, the control room emergency ventilation system will manually be placed in the chlorine isolation mode of operation to provide the required protection. The detection systems required by this speci-fication are consistent with the recommendations of Regulatory Guide 1.95, "Pro-tection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release," February 1975.

There are three toxic gas detection subsystems. The high toxic chemical concentration alarm in the Main Control Room annunciates when two of the three subsystems detect a high toxic gas concentration. An Operate/Inop keylock switch is provided for each subsystem which allows an individual subsystem to be placed in the tripped condition. Placing the keylock switch in the INOP position initiates one of the two inputs required to initiate the alarm in the Main Control Room.

~~Specified s~~**S**urveillance intervals **are determined in accordance with the Surveillance Frequency Control Program** and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

3/4.3.7.9 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE TRM.

SURVEILLANCE REQUIREMENTS (Continued)

SR 4.4.3.1.b
This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm setpoint and relative accuracy of the instrument string. ~~The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.~~

SR 4.4.3.1.c
This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. ~~The Frequency of 24 months is for a typical refueling cycle and considers channel reliability. Operating experience has proven this frequency to be acceptable.~~

SR 4.4.3.1.d
This SR provides a routine check of primary containment pressure and temperature for indirect evidence of RCS leakage. ~~Operating experience has proven this frequency to be acceptable.~~

REFERENCES

1. LGS UFSAR, Section 5.2.5.1.
2. Regulatory Guide 1.45, May 1973.
3. LGS UFSAR, Section 5.2.5.2.1.3
4. LGS UFSAR, Section 5.2.5.2.1.5
5. LGS UFSAR, Section 5.2.5.2.1.4
6. LGS UFSAR, Section 5.2.5.2.1.1(2)
7. GEAP-5620, April 1968.
8. NUREG-75/067, October 1975.
9. LGS UFSAR, Section 5.2.5.6.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. The limit of 2 gpm increase in UNIDENTIFIED LEAKAGE over a 24-hour period and the monitoring of drywell floor drain sump and drywell equipment drain tank flow rate at least once every eight (8) hours conforms with NRC staff positions specified in NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," as revised by NRC Safety Evaluation dated March 6, 1990. The ACTION requirement for the 2 gpm increase in UNIDENTIFIED LEAKAGE limit ensures that such leakage is identified or a plant shutdown is initiated to allow further investigation and corrective action. Once identified, reactor operation may continue dependent upon the impact on total leakage.

PROCEDURES AND PROGRAMS (Continued)

i. Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries For Stationary Applications,” of the following:

a. Actions to restore battery cells with float voltage < 2.13 volts, and

b. Actions to equalize and test battery cells that have been discovered with electrolyte level below the minimum established design limit.

j. Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 0.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.