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L30-86(02-05)-L
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ILLINOIS POWER COMPANY

CLINTON POWER STATION, P.O. BOX 678, CLINTON, ILLINOIS 61727

February 5, 1986

Docket No. 50-461

Director of Nuclear Reactor Regulation
Attention: Dr. W. R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Clinton Power Station
Power Ascension Program Acceleration

Dear Dr. Butler:

Illinois Power Company (IP) hereby submits the following detailed justifications pertaining to the Clinton Power Station (CPS) Power Ascension Program Acceleration (PAPA) test modifications (attached) for NRC consideration:

- Startup Test 1.0 - Chemical and Radiochemical Test Simplification
- Startup Test 3.0 - Fuel Loading Test Simplification
- Startup Test 5.1 - Control Rod Drive System/Hot Friction Testing Test Simplification
- Startup Test 5.2 - Control Rod Drive System/Hot Single Rod Scram Testing In Conjunction With Startup Test 28.0
- Startup Test 5.3 - Control Rod Drive System/Ganged Rod Testing Delete Selected Non-Essential Equipment Testing
- Startup Test 8.0 - Control Rod Sequence Exchange Test Deletion
- Startup Test 11.0 - Local Power Range Monitor Calibration Test Simplification
- Startup Test 14.0 - Reactor Core Isolation Cooling System Test Simplification
- Startup Test 18.0 - Traversing Incore Probe Uncertainty Test Deletion
- Startup Test 21.0 - Core Power-Void Mode Response Test Deletion
- Startup Test 22.0 - Pressure Regulator Test Simplification
- Startup Test 23.1 - Feedwater System Response Test Simplification
- Startup Test 24.0 - Turbine Valve Surveillance Test Simplification
- Startup Test 25.1 - Main Steam Isolation Valve Functional Tests Test Simplification
- Startup Test 26.0 - Relief Valves Test Simplification
- Startup Test 27.0 - Turbine Trip And Generator Load Rejection Test Simplification
- Startup Test 29.0 - Recirculation Flow Control System Test Simplification
- Startup Test 30.1 - Single Recirculation Pump Trip Test Simplification
- Startup Test 30.3 - Recirculation Pump Trip - Two Pumps, In Conjunction With Startup Test 27.0
- Startup Test 30.4 - Recirculation Runback In Conjunction With Startup Test 23.3

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Startup Test 30.5 - Recirculation System Cavitation Test Simplification
Test Condition 4 - Natural Circulation Operation Test Simplification

IP also requests a priority review of those test packages needed to support fuel load, i.e., the ones performed during the Open Vessel and Heatup phases. In particular, Startup Tests 1.0, 3.0, 5.1, 5.2, 5.3, 11.0, 14.0, 25.1, and 26.0 are performed during the initial two phases of startup testing and should be reviewed first. The remaining Startup Tests are performed during Test Conditions 1 through 6.

As discussed in the December 16, 1985 presentation to the Staff, the attached test packages are similar in nature to those submitted by Public Service Electric and Gas Company (PSE&G) on behalf of their Hope Creek Generating Station (HCGS). Accordingly, Attachment I provides a matrix which correlates the CPS Startup Test with the HCGS Test Number. As can be seen from Attachment I, seven packages are identical to those submitted by HCGS, ten packages are similar to HCGS with relatively minor exceptions and five are unique to CPS. Attachment II consists of the technical analysis for each of the aforementioned CPS Startup Test modifications and identifies differences, if any, between this CPS submittal and the previously submitted HCGS packages. The conclusion for each CPS item shows that the proposed modifications pose no increase in risk to the health and safety of the public and that there are no unreviewed safety questions. Upon notification of a favorable review of these packages, IP shall submit the applicable FSAR change pages for your approval.

For your information, four (4) other tests which were originally considered for inclusion in the PAPA program and discussed in the December 16 presentation have been withdrawn from consideration after detailed review by the responsible organizations. These tests were:

Startup Test 2.0 - Radiation Measurement
Startup Test 12.0 - Average Power Range Monitor Calibration
Startup Test 16.0 - Selected Process Temperatures
Startup Test 19.0 - Core Performance

Submittal of these justifications to the NRC completes this phase of the Power Ascension Test Program changes. Since these modifications impact finalizing the related power ascension testing detailed procedures, an expedited review is requested. IP is ready to meet with cognizant NRC personnel to discuss the proposed modifications should you require additional information.

Sincerely yours,

F. A. Spangenberg

F. A. Spangenberg
Manager - Licensing and Safety

EDS/ckc
Attachments
cc: B. L. Siegel, NRC Clinton Licensing Project Manager
NRC Resident Office
Regional Administrator, Region III, USNRC
Illinois Department of Nuclear Safety

ATTACHMENT I

Correlation of The Clinton Power Station (CPS) Startup Test With The Hope Creek Generating Station (HCGS) Test Number

CPS Startup Test	HCGS Test Number	Similarity to HCGS
1.0	1	Identical except not substituting Tech. Spec. surveillance.
3.0	3	Identical except for plant-specific fuel loading sequence.
5.1	-	Test simplification unique to CPS.
5.2	5	Identical except specific Test Condition 2 power level and reactor scram test.
5.3	-	Test deletion unique to CPS.
8.0	-	Test deletion unique to CPS.
11.0	9*	Identical except test deleted at heatup only.
14.0	12*	Identical except all RCIC tuning not deleted.
18.0	16	Identical except HCGS uses gamma flux detector.
21.0	19*	Identical.
22.0	20	Identical except ALF** tested at CPS.
23.1	21A	Identical.
24.0	-	Test simplification unique to CPS.
25.1	23A	Identical.
26.0	24	Identical.
27.0	25*	Identical.
29.0	27*	Identical except for ALF** and reliance on other BWR data.
30.1	-	Test simplification unique to CPS.
30.3	28B	Identical.
30.4	28D	Identical.
30.5	28E	Identical except test performed at Test Conditions 2 and 3.
Test Condition 4	Test Condition 4	Identical except all testing not deleted.

NOTE: Unlike the Public Service Electric and Gas submittal, Illinois Power Company is not submitting test modification packages for the following Startup Tests:

CPS Startup Test	HCGS Test Number
12.0	10
16.0	14A
29.0 ALF**	ALF** portion of 20, 21, and 27
25.2	23B
33.2	31
70.0	32

* The NRC has approved this power ascension test modification.
** ALF stands for Automatic Load Following.

ATTACHMENT II

STARTUP TEST 1.0 - CHEMICAL AND RADIOCHEMICAL
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 4.h and 5.a.a require chemical and radiochemical tests and measurements to demonstrate the design capability of chemical control systems to maintain reactor water quality within limits. Startup Test 1.0, Chemical and Radiochemical, demonstrates that the plant water chemistry and radiochemistry complies with these limits during the power ascension test program and also demonstrates the design capability of the plant chemistry systems. It is proposed to delete the integrated performance testing of the Reactor Water Cleanup (RWCU) System and condensate demineralizer system at Test Condition 3.

DISCUSSION:

In addition to the chemistry and radiochemistry measurements of reactor coolant and auxiliary systems, Startup Test 1.0 requires the RWCU to be taken out of service to demonstrate the integrated performance of the RWCU and condensate demineralizer systems. The no-RWCU (no-cleanup) test is planned to be performed at Test Condition 3 and 6. Although performance of this test at Test Condition 3 provides an opportunity to verify procedures and to obtain preliminary data, testing at Test Condition 3 is not required to operate at higher power levels or to be able to effectively perform the test at Test Condition 6. Testing at Test Condition 6 will demonstrate the ability of these systems to adequately manage coolant chemistry at the most demanding plant operating condition (rated power/flow).

CONCLUSION:

Performance of the no-cleanup test at Test Condition 6 provides the required data to assess the chemical control performance of the RWCU and condensate demineralizer systems. This testing satisfies the objectives of Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 4.h and 5.a.a.

Deletion of the no-cleanup test at Test Condition 3 will not adversely affect any safety systems or the safe operation of the plant since complete testing is performed at Test Condition 6 and thus does not involve an unreviewed safety question.

SIMILARITY WITH HOPE CREEK GENERATING STATION TEST NUMBER 1:

Identical in principle; however, Clinton Power Station is not proposing to substitute plant surveillance testing for portions of this startup test.

STARTUP TEST 3.0 - FUEL LOADING
TEST SIMPLIFICATION - ELIMINATE FUEL LOADING CHAMBERS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 2, establishes requirements for initial fuel loading to prevent inadvertent criticality. Requirements for continuous monit ring of the neutron flux throughout the core loading must be established. In addition, paragraph 2.a requires that the shutdown margin be verified for a partially and fully loaded core. Startup Test 3.0, Fuel Loading, provides the procedures to load fuel safely and efficiently to the full core size, and includes subcriticality checks, shutdown margin verification (including Startup Test 4.0, Full Core Shutdown Margin) and control rod functional testing (in conjunction with Startup Test 5.0, Control Rod Drive System).

Currently, it is planned to use Fuel Loading Chambers (FLC) to measure the neutron count rate during initial fuel loading because of their high sensitivity and ability to be moved near initial fuel loading and neutron sources. It is proposed to simplify the fuel loading procedure by replacing the FLC's with the Source Range Monitor (SRM) instrumentation. In addition, the startup sources will be positioned in their alternate locations (closer to the SRM detectors) and the fuel loading sequence will be modified such that initial fuel loading will begin between an SRM detector and a neutron source. Fuel loading will continue in a spiral pattern around the initial SRM until the core is fully loaded.

DISCUSSION:

Requirements during fuel loading are established to preclude inadvertent criticality. A thorough pre-fuel loading checklist is performed to verify that all systems required during fuel loading are operable. Prior to fuel loading, all control rods are verified to be fully inserted and remain fully inserted throughout the proposed fuel loading,

except during the functional/subcritical checks and the partial core shutdown margin test at which time rod movement is controlled by strict procedural guidance. Refueling interlocks prevent the withdrawal of more than one control rod at a time except during partial core shutdown margin check. No control rod movement will be performed until the minimum count rate is achieved on at least one SRM.

Predictions of the core reactivity and shutdown margin have been prepared in advance for the Clinton Power Station (CPS) fuel design and support the safe fuel loading under subcritical conditions. For the CPS initial fuel loading, it has been predicted that the effective neutron multiplication factor for a fully loaded core with all rods inserted is 0.924, and is 0.969 with the strongest worth control rod withdrawn (cold conditions). These predictions have been performed with core physics calculation methods that have been extensively qualified and have been demonstrated to be highly accurate for criticality predictions based on tests/experiments, Monte Carlo benchmark calculations and Shutdown Margin (SDM) demonstration tests for initial and reload cores (Reference 1).

In addition, rigorous Quality Assurance (QA) programs during fuel design and fuel and control blade manufacturing ensure that the fuel and control blades are manufactured as specified. These QA programs (described in Reference 2) apply quality system elements necessary to provide assurance that systems and components meet the quality requirements of applicable codes, standards and regulatory agency requirements. General Electric's QA program has been reviewed by the NRC and found to comply with all applicable requirements of Appendix B to 10CFR50 (Reference 3). As a result, these pre-fuel loading precautions and measures provide significant assurances against inadvertent criticality during initial fuel loading.

For the proposed fuel loading procedure, the fuel will be initially loaded between a source at its alternate location and the closest SRM. Figure 1 shows the proposed fuel loading sequence. Because of the fixed location of the SRM's and the distance from the sources, the SRM count rate will initially be less than 0.7 counts per second (cps), the Plant Technical Specification minimum SRM count rate. As fuel is loaded,

neutronic coupling between the source and detector will occur and the count rate will increase. The estimated minimum SRM count rate with 16 bundles loaded as a function of the source strength has been determined based on past BWR startup test results from seven plants using FLC's. SRM count rates were also available for one of the seven plants. The results are presented in Figure 2. Figure 2 shows that, with appropriate conservatisms, a source strength of 500 curies is required to assure that the minimum count rate of 0.7 cps is achieved with the initial 16 bundles loaded. The SRM data, for a startup where the alternate source location was used (source strength of 1855 curies), indicated an SRM count rate of 12 cps with six bundles loaded and 30 cps with 16 bundles loaded. Similar performance is expected during fuel loading at CPS. Nevertheless, an exemption to the Technical Specification requirement for a minimum count rate (0.7 cps) during the loading of the initial 16 bundles is required. Since the SRM system is not safety related and no credit is taken for the SRM's in the safety analysis for these conditions, this exemption will not adversely affect any safety systems or the safe operation of the plant.

To further support this exemption, a core reactivity calculation was performed for the initial 16 bundles loaded to demonstrate that even with all of the control rods withdrawn, the partial loading would remain subcritical with significant margin. The fuel bundle types and loading configuration of the 16 bundles were based on the proposed fuel loading procedure of Figure 1 and are shown in Figure 3. For a moderator temperature of 20°C, the resulting effective neutron multiplication factor was 0.977. This analysis demonstrates that the initial 16 bundle loading will remain subcritical even if the control rods are withdrawn by 2.3%Δk, thus further assuring that the SRM monitoring requirements can be exempted during this portion of the fuel loading procedure. Similar exemptions of SRM monitoring requirements (for fewer bundles) during fuel loading have been approved by the NRC for several reload licenses (References 4, 5 and 6).

After the initial 16 bundles are loaded and an SRM is on scale, the fuel loading will continue in a spiral fashion as shown in Figure 1. Since only one SRM will initially be on scale, a portable source will be used to periodically demonstrate operability of the SRM's located in areas with no fuel. One of the SRM's will be required to maintain continuous visual indication in the control room until other SRM's are on scale. Use of a portable source to demonstrate operability of the remaining SRM's has previously been approved by the NRC (References 4 and 6). The portable source is widely used in the nuclear industry as a bugging source for detector calibration and is an easy device to operate with no complex or unsafe maneuvers required.

When 144 fuel bundles have been loaded (Step 36 in Figure 1), two SRM's will be surrounded by fuel and indicating greater than 0.7 cps. At this time, a partial core shutdown margin test will be performed as required by Regulatory Guide 1.68. The partial core configuration (fuel bundle types) for this off-center loading is almost identical to the partial core configuration resulting from a standard center spiral loading . Core physics calculations for the off-center partial SDM test will be performed for the proposed fuel loading sequence. After the partial core SDM test, the remaining fuel will be loaded based on the sequence shown in Figure 1 until the core is fully loaded. During this portion of the fuel loading, at least two SRM's will be operable, with at least one providing continuous visual indication in the control room. SRM's not surrounded by fuel will be periodically checked for operability using the portable source. Once the core is fully loaded, the full core SDM test will be performed using the standard procedures. Attachment 1 provides a summary of the major fuel loading steps.

Currently, Plant Technical Specifications require that at least two SRM's are operable and continuously visible in the control room. The SRM's are required to be located in the quadrant where fuel loading occurs and in an adjacent quadrant. For the proposed fuel loading procedure, a Special Test Exception (3/4.10.6, Special Instrumentation - Initial Core Loading) is required for the Technical Specifications to

incorporate the changes to the flux monitoring requirements (Attachment 2). Initially, continuous visual indication and minimum count rate requirements for the SRM's are exempted for the first 16 fuel bundles loaded since the core will be subcritical even with all control rods withdrawn. Two SRM's, one in the quadrant where fuel loading is taking place and one in an adjacent quadrant, will still be required to be operable, but one will be allowed to be demonstrated operable by using a portable neutron source. This operability check will be periodically performed for those SRM's located in areas with no fuel loaded. As stated before, this method of demonstrating SRM operability has been previously approved for other BWR's (References 4 and 6).

The SRM system is not safety related and no credit is taken for the SRM's under these conditions (fuel loading with control rods inserted, i.e., core subcritical) in the safety analysis of accidental positive reactivity insertions. The SRM's provide indication of neutron flux changes as a matter of good practice during fuel loading and thus no safety requirements exist for an SRM detector. Therefore, continuous visual indication of only one SRM does not adversely affect any safety systems or the safe operation of the plant since the core will be subcritical during fuel loading and at least two SRM's will be demonstrated operable prior to and during fuel loading.

CONCLUSION:

By locating the startup neutron sources in alternate positions and modifying the fuel loading sequence, the SRM's will be capable of providing continuous monitoring of the neutron flux throughout the loaded core after the minimum count rate is initially achieved. Monitoring requirements and subcriticality checks during loading of the initial fuel bundles (up to 16 bundles) can be exempted because even with all of the control rods withdrawn, this configuration would be subcritical because of the high neutron leakage. Pre-fuel load activities ensure that systems required for fuel loading are operable, control rods are inserted during the fuel loading except during special

controlled tests, and core reactivity predictions aid in the evaluation of the measured responses during fuel loading. Together, these procedures and precautions provide assurance against inadvertent criticality. Since the safety analysis does not require a minimum count rate on the SRM system under these conditions, the proposed change in the fuel loading procedure will not adversely affect any safety system or safe operation of the plant and therefore does not involve an unreviewed safety question. Startup Test 3.0, Fuel Loading, can therefore be simplified by replacing the Fuel Loading Chambers with SRM's and using an off-center spiral loading sequence.

SIMILARITY WITH HOPE CREEK GENERATING STATION TEST NUMBER 3:

Identical; except for plant-specific fuel loading sequence.

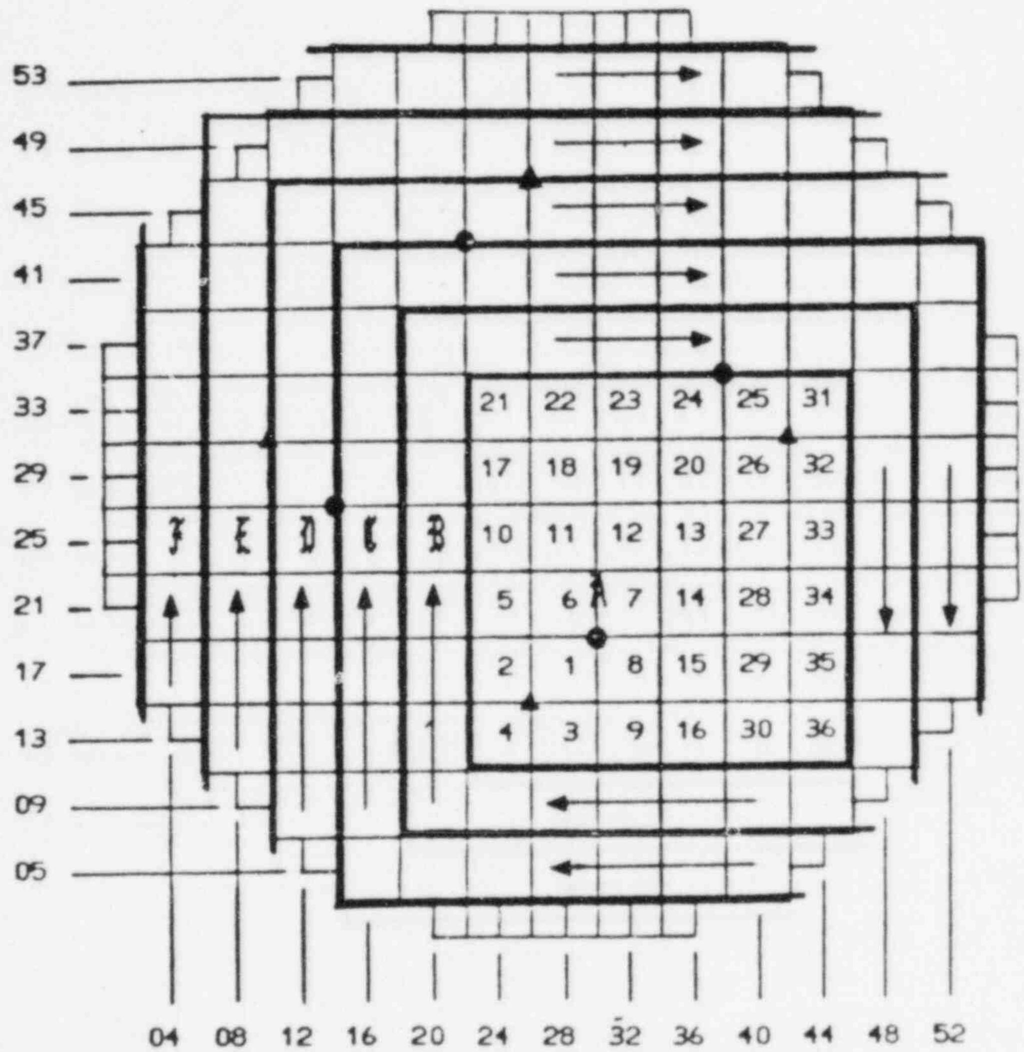
REFERENCES:

1. "Steady State Nuclear Methods", General Electric Company Licensing Topical Report, May 1985 (NEDO-30130-A).
2. "Nuclear Energy Business Operation Quality Assurance Program Description," General Electric Company, March 1985 (NEDO-11209, Revision 5).
3. Letter, G.G. Zech (NRC) to J.M. Case (GE), "NRC Acceptance of Revised General Electric Quality Assurance Topical Report," April 19, 1985.
4. Amendment No. 27 to Facility Operating License No. DPR-63, Niagara Mohawk Power Co., Nine Mile Point Station Unit No. 1, Docket No. 50-220, March 2, 1979.
5. Amendment No. 66 to Facility Operating License No. DPR-57, Georgia Power Co. et. al., Edwin I. Hatch Nuclear Station Unit No. 1, Docket No. 50-321, June 12, 1979.
6. Amendment No. 5 to Facility Operating License No. NPF-29, Grand Gulf Nuclear Station Unit No. 1, Docket No. 50-416, October 12, 1985.

ATTACHMENTS:

1. Summary of Major Fuel Loading Steps
2. Technical Specification Change - Special Test Exception 3/4.10.6

Figure 1 - Proposed Fuel Loading Sequence



▲ = Alternate Source Location
● = SRM Detector Location
(Note: Letters denote fuel loading regions)

Figure 2 - SRM Count Rate Estimate

Startup Test Data

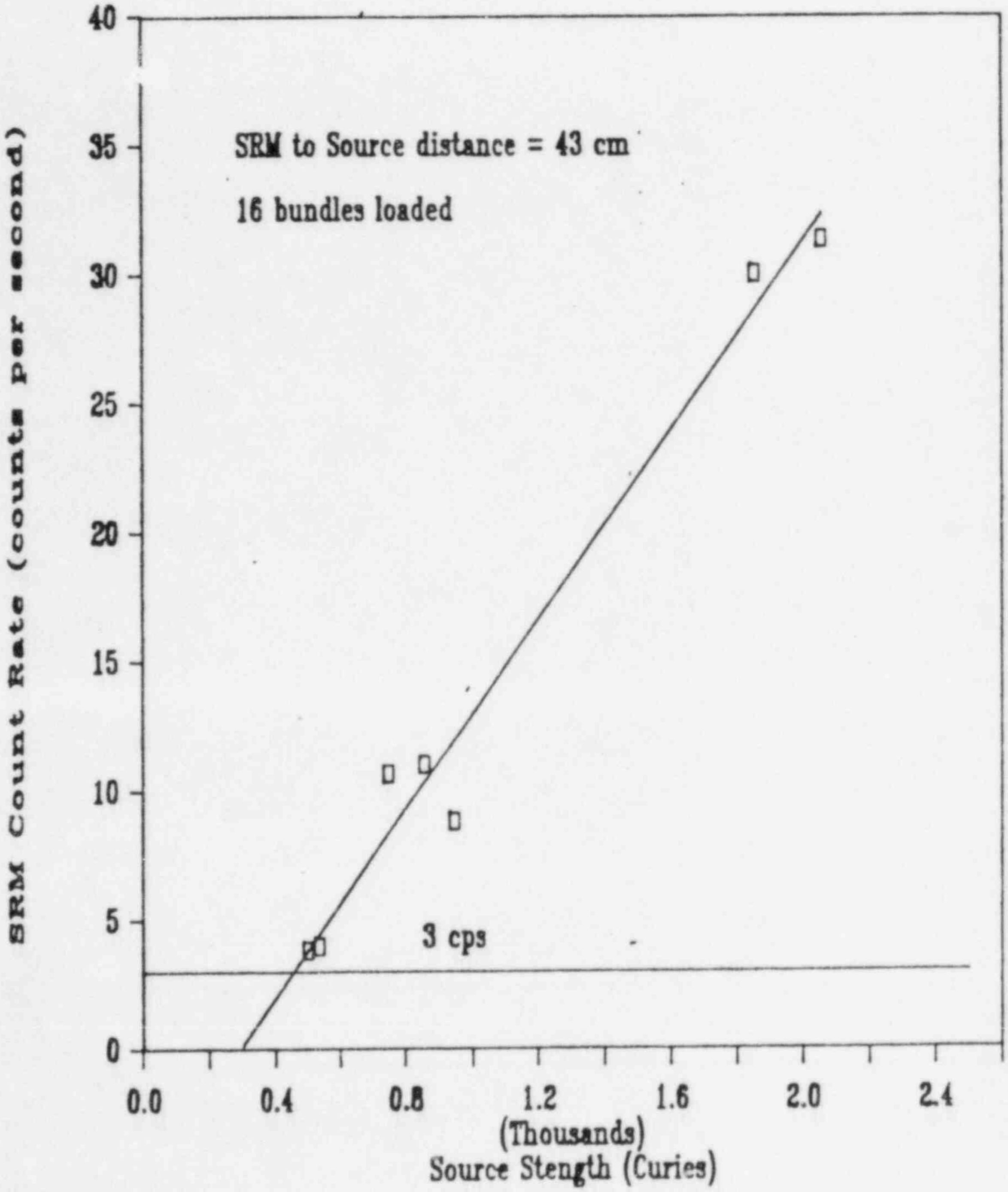
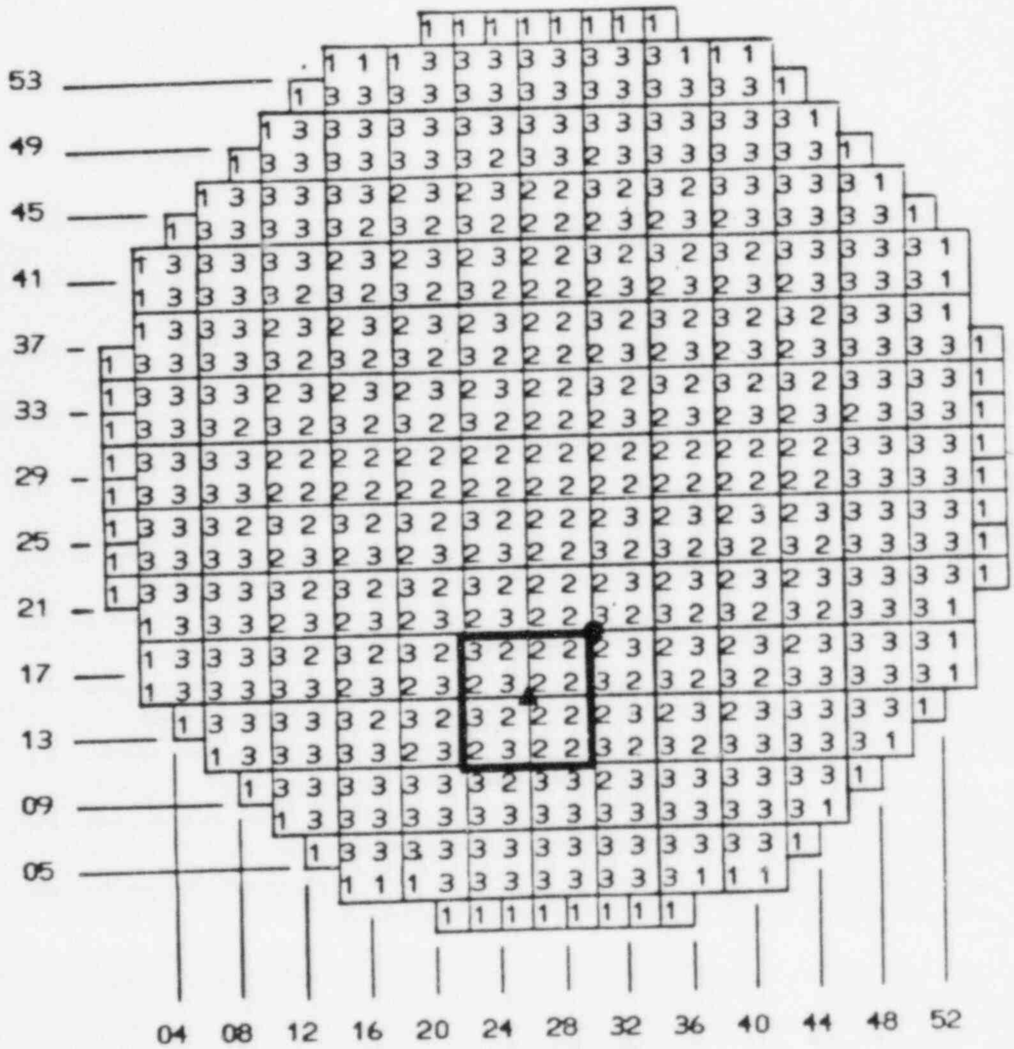


Figure 3 - Fuel Loading Pattern



1 = 0.71 w/o U235 (76)
2 = 1.54 w/o U235 (188)
3 = 2.00 w/o U235 (360)
▲ = Alternate Source Location
● = SRM Detector

□ = Analyzed Region

Attachment 1 - Summary of Major Fuel Loading Steps

1. Prerequisites

   a. Neutron sources have been loaded into the alternate source locations.

   b. Nuclear instrumentation (SRM's and IRM's) has been installed and activated in the design configuration.

   c. The "shorting links" have been removed from the Reactor Protection System (RPS) and the non-coincident scram setpoints have been set.

   d. All control rods are fully inserted and have been functionally tested within approximately two weeks prior to the start of fuel loading and have been scram tested within approximately six weeks prior to fuel loading.

   e. Fuel support castings are installed and locations verified to assure proper orificing.

   f. The operability of the SRM's has been verified.

   g. All other systems required to be operable by Technical Specifications shall be determined to be operable.

   h. The reactor mode switch is locked in the REFUEL/SHUTDOWN mode during fuel loading.

Attachment 1 - Summary of Major Fuel Loading Steps (cont'd)

2. Procedure

a. Begin fuel loading following the steps shown in Figure A-1. (Off-center loading).

b. Fuel loading begins at cell location 28,17 and continues until the first 16 bundles have been loaded (Steps 1-4, Figure A-1).

c. Perform SRM operability check for SRM adjacent to the initial 16 bundles (30,19). Perform channel checks for other SRMs according to Technical Specification requirements. A portable source is to be used for this purpose when an SRM is not surrounded by fuel.

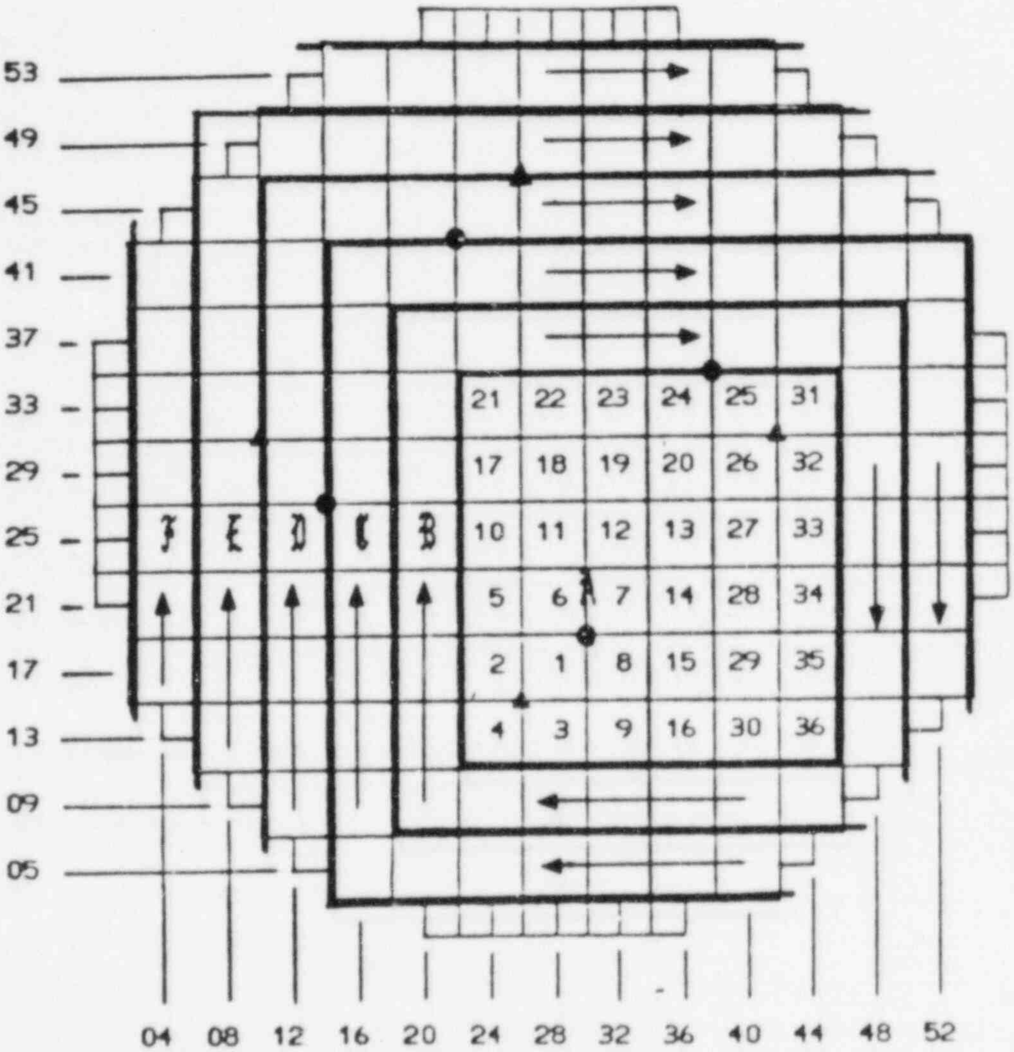
d. Continue fuel loading according to Figure A-1 until Region A is completely loaded (144 bundles). During the process of fuel loading, SRM(s) count rate should be recorded. This is used to generate 1/M plots. Perform control rod functional checks and subcritical tests when the minimum count rate is achieved on at least one SRM channel.

e. Perform the partial core shutdown margin test after 144 bundles are loaded (Region A, Figure A-1). Core physics calculations will be performed to support this test.

f. Continue fuel loading in Region B. The loading is continued according to sequential order of regions (Figure A-1) until the core is fully loaded.

g. As soon as each SRM is on scale (0.7 cps or higher) during loading, SRM calibration is performed and documented.

Figure A-1 - Proposed Fuel Loading Sequence



▲ = Alternate Source Location
● = SRM Detector Location
(Note: Letters denote fuel loading regions)

Attachment 2 - Technical Specification Change

SPECIAL TEST EXCEPTION (NEW) DRAFT

3/4.10.6 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

LIMITING CONDITION FOR OPERATION

3.10.6 During initial core loading within the Startup Test Program the provisions of Specification 3/4.9.2 may be suspended provided that at least two source range monitor (SRM) channels with detectors inserted to the normal operating level are OPERABLE with:

a. One of the required SRM channels continuously indicating* in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant,**

c. The RPS "shorting links" shall be removed prior to and during fuel loading,

d. The reactor mode switch is OPERABLE and locked in the REFUEL/SHUTDOWN position.

APPLICABILITY OPERATIONAL CONDITION 5

ACTION

With the requirements of the above specification not satisfied, immediately suspend all operations involving initial core loading.

SURVEILLANCE REQUIREMENTS

4.10.6.1 Within one hour prior to and at least once per 12 hours during the initial core loading verify that:

a. The above required SRM channels are OPERABLE by:

1. Performance of a CHANNEL CHECK***

2. Confirming that the above required SRM detectors are at the normal operating level and located in the quadrants required by Specification 3.10.6.

3/4.10.6 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING (cont'd) DRAFT

SURVEILLANCE REQUIREMENTS (cont'd)

4.10.6.1 (cont'd)

b. The RPS "shorting links" are removed.

c. The reactor mode switch is locked in the REFUEL position.

4.10.6.2 Perform a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start and at least once per 7 days during initial core loading.

4.10.6.3 For at least one SRM channel, verify that the count rate is at least 0.7 cps****:

a. Immediately following the loading of the first 16 fuel bundles.

b. At least once per 12 hours thereafter during initial core loading.

* Up to 16 fuel bundles may be loaded without a visual indication of count rate.

** The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

*** Check may be performed by use of movable neutron source.

**** Provided signal-to-noise is $\geq$ 2. Otherwise, 3 cps.

REFUELING OPERATIONS DRAFT

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:

a. Continuous visual indication in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

c. Unless adequate shutdown margin has been demonstrated, the "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn.

APPLICABILITY: OPERATIONAL CONDITION 5.**

ACTION:

With the requirements of the above specification not satis[illegible]ed, immediately suspend all operations involving CORE ALTERATIONS and insert a[illegible] insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

1. Performance of a CHANNEL CHECK,
2. Verifying the detectors are inserted to the normal operating level, and
3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

**See Special Test Exception 3.10.6

STARTUP TEST 5.1 - CONTROL ROD DRIVE SYSTEM/HOT FRICTION TESTING
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 2.b requires friction testing (for BWR's) of control rods after the core is fully loaded. Startup Test 5.1, Control Rod Drive System, performs friction testing of all control rod drives (CRD) at both hot zero power and cold conditions for BWR/6 plants. It is proposed to reduce the number of CRD's to be hot friction tested to four.

DISCUSSION:

Performance of the control rod drives during friction testing is compared to acceptance criteria which require that during continuous insertion, the differential pressure variation for the CRD must not exceed a specified limit. If the limit is exceeded during continuous insertion, a settling test is performed to determine the differential settling pressure and variation.

CRD friction testing at the cold condition satisfies the objectives of Regulatory Guide 1.68, Appendix A, paragraph 2.b. The additional testing of Startup Test 5.1 at hot zero power is not required by the regulations and is often impractical to perform at these conditions. The hot condition represents a safety hazard to the technicians that operate the instrumentation required for friction testing.

Out of the 777 CRD's at the five BWR/6 plants which have completed startup tests (145 at Cofrentes, 193 at Grand Gulf-1, 145 at Kuo Sheng-1, 145 at Kuo Sheng-2, and 149 at Leibstadt) none have failed the hot friction test. Using zero failures out of 777, the Poisson distribution provides that there is 95% confidence that the expected number of CRD's which would fail the hot friction test is less than 0.56 for a BWR/6 plant with 145 CRD's (i.e., Clinton Power Station).

Therefore, it is unneccessary to friction test all the CRD's at hot zero power and it is recommended that only four CRD's be hot friction tested to assure that there is no widespread problem because of thermal expansion. The four CRD's chosen should be the same as those for which scram timing is to be repeated during plant heatup. Plants prior to the BWR/6 product line have typically performed hot friction tests on only four CRD's.

CONCLUSION:

CRD friction testing at the cold condition during Startup Test 5.1 demonstrates the acceptable performance of the CRD system and satisfies the objectives of Regulatory Guide 1.68, Appendix A, paragraph 2.b for friction testing. The proposed change to reduce the number of CRD's to be hot friction tested does not adversely affect any safety systems or safe operation of the plant and therefore does not involve an unreviewed safety question. Startup Test 5.1, Control Rod Drive System, can therefore be simplified by reducing the number of CRD's to be hot friction tested to four.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 5:

This test simplification is unique to Clinton Power Station. However, the resultant testing is consistent with currently planned HCGS testing.

STARTUP TEST 5.2 - CONTROL ROD DRIVE/HOT SINGLE ROD SCRAM TESTING IN CONJUNCTION WITH TEST NUMBER 28.0

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 2.b, 4 and 5.h require that control rod scram testing be performed at plant conditions that bound those under which the control rods might be required to function to achieve plant shutdown. The control rods are required to scram within the specified times specified by Plant Technical Specifications and assumed in Final Safety Analysis Report (FSAR) analysis. Startup Test 5.2, Control Rod Drive System, demonstrates the performance of the Control Rod Drive (CRD) System over the full range of reactor operating conditions. It is proposed to delete scram time testing of all but four CRD's at rated reactor pressure during Test Condition Heatup and to obtain the rated reactor pressure scram data from a planned full core scram at Test Condition 2 (approximately 10-25% power).

DISCUSSION:

Scram performance of the control rod drive system is compared to acceptance criteria which require that the control rods scram within a specified time. Control rod drive scram timing tests are begun during preoperational testing and all later testing is an extension of the preoperational program. Following fuel loading, with the reactor in the cold shutdown condition, all CRD's are individually withdrawn and scram time tested. This test provides a data base from which four control rods are selected with scram times among the longest of those measured or which exhibit unusual operating characteristics. These four CRD's are individually scram time tested during heatup at approximately 600 and 800 psig reactor pressure.

Currently, at rated reactor pressure and low power level, all of the CRD's are again scram time tested. From this data base, four CRD's are selected for additional individual testing and for monitoring during full reactor scrams during the power ascension test program. It is proposed to only test four rather than all of the CRD's at rated reactor pressure during Test Condition Heatup. Scram times for the remaining CRD's would be obtained from the planned reactor scram during

performance of the Shutdown from Outside the Control Room, Startup Test 28.0, at Test Condition 2 (approximately 10-25% power). This full reactor scram would provide scram times for all CRD's except those which were not at a fully withdrawn position prior to the scram. Four CRD's would be selected for monitoring during future full reactor scrams.

Deferment of testing for all the CRD's from rated reactor pressure/low power to about 10-25% power is considered acceptable because (1) all CRD's are scram time tested at cold conditions, and (2) four CRD's are monitored during reactor heatup to rated pressure to determine the effects of increasing temperature and pressure. The combination of this data provides an indication of the performance of all CRD's at rated temperature and pressure. Increasing reactor power level should have little effect on CRD scram times. However scram times for fully withdrawn control rods would be measured during the first planned, full reactor scram. In the event of an inadvertent reactor scram, recording equipment will automatically start and provide a record of CRD scram performance.

Scram time measurements at rated reactor pressure will not be performed for 29-32 CRD's that are not fully withdrawn during Startup Test 28.0 at Test Condition 2. This is justified because analysis shows that in the extreme case of 32 CRD's plus an additional eight inoperable CRD's not being inserted during a scram (see Figure 1 for CRD pattern), the effect on negative reactivity insertion is small (see Figure 2). This is attributed to the even distribution of the 32 A sequence CRD's which are assumed to not scram. As a result, the impact on FSAR safety analyses is negligible. Analyses have been performed and demonstrate that the peak heat flux during a pressurization event does not change when the 32 CRD's are assumed to not scram.

Plant Technical Specifications would be changed to exempt up to 32 control rods (all within the A or B sequence and separated by at least one control rod in all directions including the diagonal) from scram

time testing. Although the above approach would exempt the 32 CRD's from scram time measurements at rated reactor pressure, insertion indication is provided for these CRD's during all full reactor scrams. Therefore, full insertion can be verified and estimates of insertion times can be obtained from the data recorded during the performance of Startup Test 28.0 at Test Condition 2.

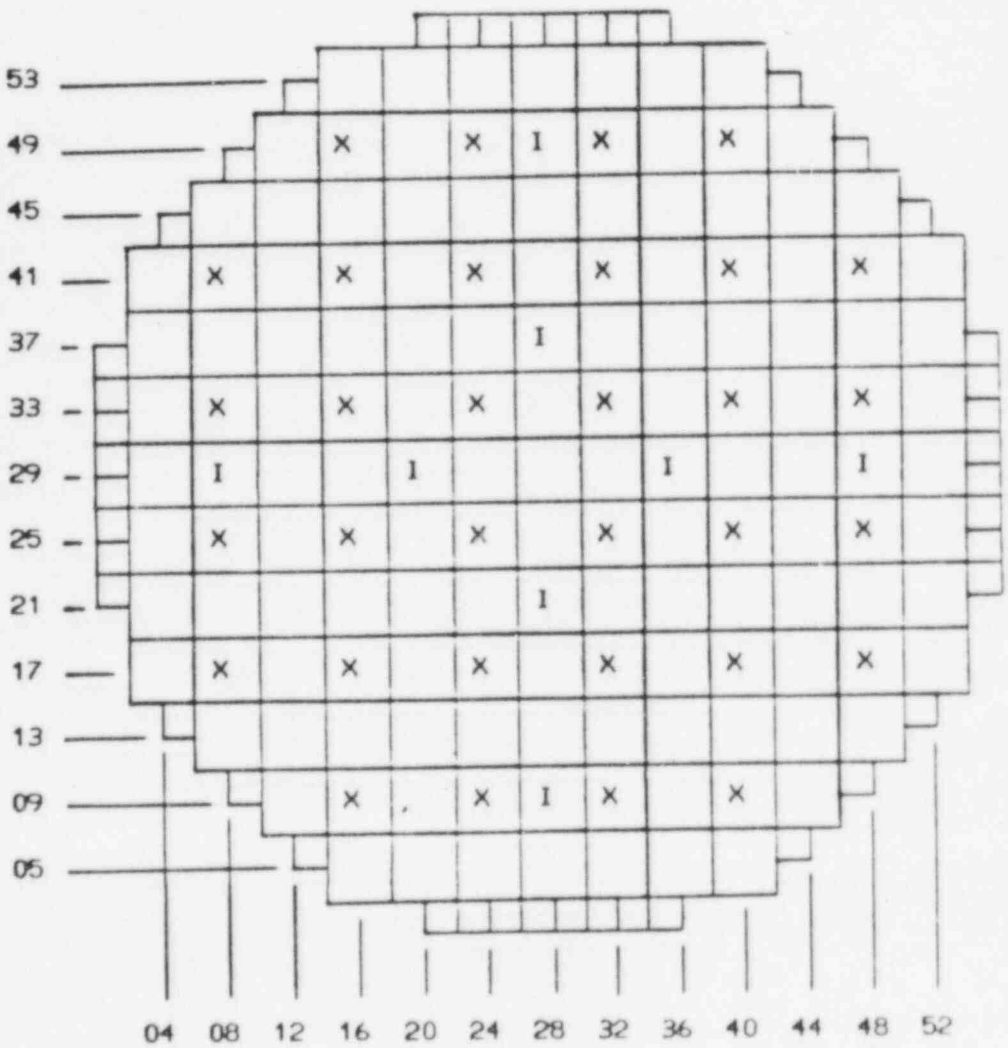
CONCLUSION:

Compensatory actions such as scram time testing of four CRD's during heatup to show the effect of increasing reactor temperature and pressure, obtaining insertion times during a full reactor scram, and performing the safety analysis neglecting the scram reactivity contribution from 32 CRD's, provide sufficient assurance that safety margins are maintained with power ascension testing changed to delete scram time testing of all but four CRD's at rated pressure during Test Condition Heatup and to exempt 32 CRD's from scram time testing at hot conditions for the initial fuel cycle. The proposed testing change will not adversely affect any safety systems or safe operation of the plant. In addition, since the safety analysis has been performed neglecting the scram reactivity contribution from the 32 CRD's,the change does not involve an unreviewed safety question.

SIMILARITY WITH HOPE CREEK GENERATING STATION TEST NUMBER 5:

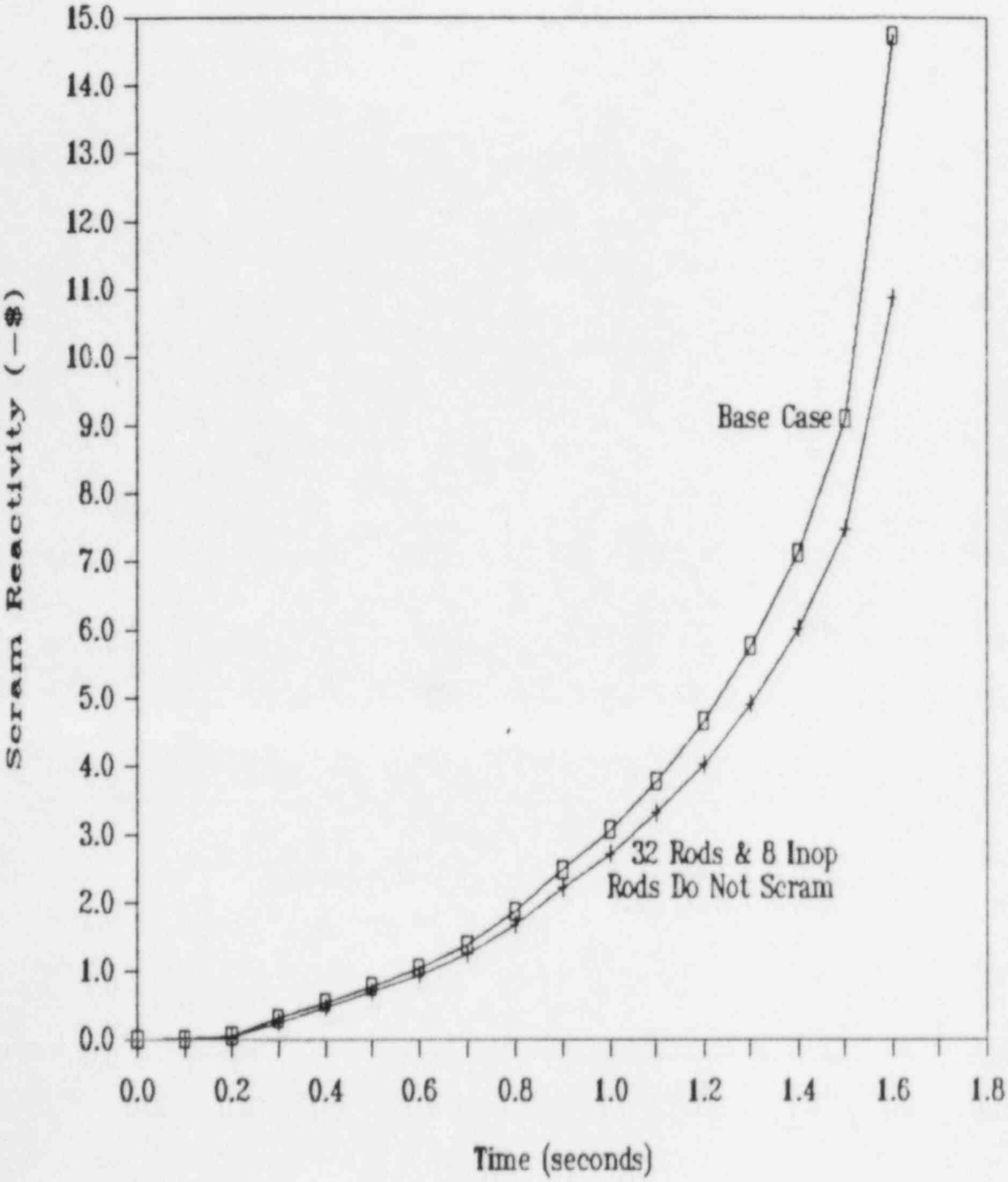
Identical in principle; the differences are: (1) the power level at which Test Condition 2 tests (approximately 10-25% power) are performed and (2) the specific test during which the planned reactor scrams to obtain scram times are performed.

Figure 1 - Control Rod Scram Distribution



X = 32 Exempted Control Rods
I = 8 Inoperable Control Rods

Figure 2 - Control Rod Drive Hot Scram Reactivity



REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES DRAFT

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod* from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed the following limits:

Reactor Vessel Dome Pressure (psig) **	Maximum Insertion Times to Notch Position (Seconds) 43	29	13
950	0.31	0.81	1.44
1050	0.32	0.86	1.57

APPLICABILITY: OPERATIONAL CONDITION 1 and 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Surveillance requirement 4.1.3.2.a or b, operation may continue provided that:

1. For all "slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, the individual scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig) **	Maximum Insertion Times to Notch Position (Seconds) 43	29	13
950	0.38	1.09	2.09
1050	0.39	1.14	2.22

2. For "fast" control rods, i.e., those which satisfy the limits of Specification 3.1.3.2, the average scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig) **	Maximum Insertion Times to Notch Position (Seconds) 43	29	13
950	0.30	0.78	1.40
1050	0.31	0.84	1.53

* For the initial fuel cycle only, up to 32 control rods, not occupying adjacent locations in any direction, including the diagonal, may be exempted from scram time testing under hot, pressurized conditions, provided they meet all other surveillance requirements, and that all scram reactivity requirements of the plant safety analysis are met with no scram contribution from these rods.

** For intermediate reactor vessel dome pressure, the scram time criterion is determined by linear interpolation at each notch position.

REACTIVITY CONTROL SYSTEMS   DRAFT

LIMITING CONDITION FOR OPERATION Continued

ACTION: (Continued)

3. The sum of "fast" control rods with individual scram insertion times in excess of the limits of ACTION a.2 and of "slow" control rods does not exceed 5.

4. No "slow" control rod, "fast" control rod with individual scram insertion time in excess of the limits of ACTION a.2, or otherwise inoperable control rod occupies an adjacent location in any direction, including the diagonal, to another such control rod.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. With a "slow" control rod(s) not satisfying ACTION a.1, above:

1. Declare the "slow" control rod(s) inoperable, and

2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more "slow" control rods declared inoperable.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

c. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Specification 4.1.3.2.c, operation may continue provided that:

1. "Slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, do not make up more than 20% of the 10% sample of control rods tested.

2. Each of these "slow" control rods satisfies the limits of ACTION a.1.

3. The eight adjacent control rods surrounding each "slow" control rod are:

   a). Demonstrated through measurement within 12 hours to satisfy the maximum scram insertion time limits of Specification 3.1.3.2, and

   b). OPERABLE.

4. The total number of "slow" control rods, as determined by Specification 4.1.3.2.c, when added to the sum of ACTION a.3, as determined by Specification 4.1.3.2.a and b, does not exceed 5.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

Attachment 1 (cont'd)

REACTIVITY CONTROL SYSTEMS DRAFT

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods* shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS*** or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

* For the initial fuel cycle only, up to 32 control rods, not occupying adjacent locations in any direction, including the diagonal, may be exempted from scram time testing under hot, pressurized conditions, provided they meet all other surveillance test requirements, and that all scram reactivity requirements of the plant safety analysis are met with no scram contribution from these rods.

** Except movement of SRM, IRM, or special removable detectors or normal control rod movement.

STARTUP TEST 5.3 - CONTROL ROD DRIVE SYSTEM/GANGED ROD TESTING
DELETE SELECTED NON-ESSENTIAL EQUIPMENT TESTING

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2; August 1978), Appendix A, paragraph 2.b requires that testing of the control rod withdrawal and insert speeds, sequencers and control functions be tested after the core is fully loaded. Startup Test 5.3, Control Rod Drive System, performs testing of the Control Rod Drive (CRD) System over a wide range of operating conditions to satisfy the objectives of Regulatory Guide 1.68, Appendix A, paragraph 2.b. In particular, testing is performed on the ganged rod mode of operation for the Clinton Power Station (CPS). This testing is performed at cold conditions after the fuel is loaded. It is proposed to delete testing of the ganged rod mode of operation of the CRD system because it will have been tested during the preoperational phase of the testing program.

DISCUSSION:

Performance of the ganged rod mode of operation of the CRD system is compared to acceptance criteria which require that the control rods within a gang move together such that all rods are within two notches of all other rods in the gang. The ganged mode of operation of the CRD system is not a safety related function but is instead an operational improvement. Testing of the ganged rod mode will be performed during the preoperational phase of testing to demonstrate compliance with the acceptance criteria. Since the ganged rod mode is not safety related, no additional testing is required.

CONCLUSION:

The ganged rod mode of operation will have been testing during preoperational testing. Independently, the systems which enforce control rod sequences for compliance with safety analysis (control rod drop accident and rod withdrawal error) are tested to demonstrate compliance with their system requirements, and as such, the change does not involve an unreviewed safety question. Therefore, the ganged rod mode does not need to be tested during the power ascension program.

SIMILARITY WITH HOPE CREEK GENERATING STATION TEST NUMBER 5:

This test deletion is unique to CPS.

STARTUP TEST 8.0 - CONTROL ROD SEQUENCE EXCHANGE
JUSTIFY TEST DELETION

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2; August 1978), Appendix A, paragraph 5.c requires that the licensee "demonstrate that core limits will not be exceeded during or following exchange of control rod patterns that will be permitted during operation (the demonstration test should be conducted at the highest power level at which control rod pattern exchanges will be allowed during plant operation)." Startup Test 8.0, Control Rod Sequence Exchange, performs a representative sequence exchange of control rod patterns at a significant power level. It is proposed to delete Startup Test 8.0, and alternatively, adhere to applicable generic sequence exchange procedures when required.

DISCUSSION:

Rod patterns will be periodically exchanged during plant operations to more nearly equalize fuel assembly exposures. These sequence exchanges are typically performed every 1000 MWD/st in core average exposure. The control rod sequence exchange begins on the 100 percent load line by reducing core flow to minimum and reducing thermal power to between the low power setpoint of the rod pattern control system and the thermal power necessary to keep nodal powers below the PCIOMR threshold. The control rod sequence exchange is performed a row or column at a time, starting at one side of the core and working row by row or column by column across the entire width of the core.

The purpose of the control rod sequence exchange test during the power ascension program is to assure that several objectives are met. Such objectives can be met by the use of generic procedures. Regulatory Guide 1.68 requires a demonstration that core limits (e.g., MCPR, MAPLHGR, MLHGR) will not be exceeded during the sequence exchange. Performing a sequence exchange establishes an asymmetric state of the core. Core calculations performed by the process computer during an asymmetric core state involve higher uncertainties than calculations performed during a symmetric state. General procedures are available

which do not depend on the asymmetric calculations performed by the process computer. Currently recommended sequence exchange procedures (Reference 1) ensures sufficient margin to core limits, such that the exchange procedures themselves assure that core limits will not be exceeded during the exchange. These procedures have been used to successfully perform sequence exchanges. Backup 3-D analytical calculations (that do not require core symmetry) established that a large margin was maintained to core limits during the exchange. Therefore, the generic procedures assure that core limits are not exceeded during sequence exchanges at power. During startup, although a representative sequence exchange using the generic procedures may optionally be performed for the purpose of familiarizing the plant operating and technical staff with the operation of the facility, the test is not required to further qualify the generic procedures.

CONCLUSION:

Control rod sequence exchanges are performed using generic procedures that have been demonstrated at operating plants. Sufficient actual reactor data and analytical back up calculations along with the wide margin to core limits that must be established before starting the exchange, give adequate confidence that the procedures are generically applicable and do not require a qualification test for the startup of each new plant. These generic procedures therefore, satisfy the objectives of Regulatory Guide 1.68 (Revision 2; August 1978), Appendix A, paragraph 5.c, and a demonstration during startup testing that implicitly demonstrates the acceptability of the sequence exchange procedures is not required. Therefore, Startup Test 8.0, Control Rod Sequence Exchange, can be deleted from the power ascension program. This change will not adversely affect any safety systems or the safe operation of the plant and thus does not involve an unreviewed safety question.

SIMILARITY WITH HOPE CREEK GENERATING STATION TEST NUMBER 6:

This test deletion is unique to Clinton Power Station.

REFERENCES:

1. "Preconditioning Interim Operating Management Recommendations (PCIOMR) Implementation Procedures," General Electric Company Proprietary, February 1982 (NEDE-21493, Revision 5).

STARTUP TEST 11.0 - LOCAL POWER RANGE MONITOR CALIBRATION
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 5.y requires that incore neutron flux instrumentation used to calculate thermal power level be calibrated as required and its operation verified. Startup Test 11.0, Local Power Range Monitor Calibration, performs the required calibration of the Local Power Range Monitor (LPRM) instrumentation. In addition, the test currently plans to verify LPRM response during control rod movement at Test Condition Heatup. It is proposed to delete the heatup testing of the LPRM response.

DISCUSSION:

Calibration of the LPRM system at several test conditions satisfies the objectives of Regulatory Guide 1.68, Appendix A, paragraph 5.y. The additional testing of Startup Test 11.0, to verify the LPRM response during heatup is not required by the Regulatory Guides and is often impractical to perform at these conditions. During heatup, the neutron flux level is low enough such that a large number of the LPRM detectors have signals which are less than the downscale value. Therefore, indication of the LPRM response is not available for the operator to verify the response. During control rod movement at power levels where sufficient indication of LPRM signals exists, indications of LPRM response can readily be observed by the operator. This response is often used as a confirmatory indication of control rod movement and provides sufficient demonstration of the LPRM system response when combined with the required calibration procedures. In addition, the process computer program, OD-8, can be used to verify the LPRM readings with the Traversing Incore Probe (TIP) predicted power shape.

CONCLUSION:

LPRM calibration during Startup Test 11.0 demonstrates the acceptable performance of the LPRM system and satisfies the objectives of Regulatory Guide 1.68, Appendix A, paragraph 5.y. The proposed change to delete the verification of LPRM response does not adversely affect any safety systems or safe operation of the plant and therefore does not involve an unreviewed safety question. Startup Test 11.0, Local Power Range Monitor Calibration, can therefore be simplified by deleting the LPRM response checks during heatup.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS)TEST NUMBER 9:

Identical in principle; however, the proposed change at Clinton Power Station is to delete this test at Test Condition Heatup rather than at Test Conditions Heatup and 1 as at HCGS.

STARTUP TEST 14.0 - REACTOR CORE ISOLATION COOLING SYSTEM
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 4.k and 5.k require the demonstration of the operability of steam-driven engineered safety features. Startup Test 14.0, Reactor Core Isolation Cooling System, verifies the proper operation of the Reactor Core Isolation Cooling (RCIC) System over its required operating pressure range. Testing is performed at low reactor pressures and near rated reactor pressures during Test Condition 1. Testing of the RCIC is performed by flow injection into a test line leading to the RCIC storage tank (RST) and by flow injection directly into the reactor vessel. Currently, tuning of the RCIC controllers is performed at both low pressures and near rated reactor pressure. It is proposed that the controller tuning at low reactor pressures be minimized.

DISCUSSION:

Response of the RCIC system is determined by analyzing test data and comparing to acceptance criteria which define the required performance of the system. The criteria place limits on the RCIC flow response times, RCIC turbine dynamic response and speed and flow control loop stability. In addition, it is required that the RCIC turbine does not trip or isolate during automatic or manual start tests. Testing is performed at low reactor pressures and near rated reactor pressure to demonstrate compliance with these criteria over the expected range of operating conditions. Initially, a set of RST injection tests are performed with manual and automatic starts at 150 psig and near rated reactor pressure conditions. This RST testing is done to demonstrate the general system operability and for making most controller adjustments. Experience at previous plants has shown that controller tuning at the low pressure condition does not result in optimum performance at higher pressures.

Reactor vessel injection tests follow to complete the controller adjustments and to demonstrate automatic starting from a "cold" standby condition ("cold" is defined as a minimum of 72 hours without any kind of RCIC operation). During startup testing, the turbine speed controller will be tuned first, near rated reactor pressure in the RST injection mode. Since the reactor pressure vessel injection testing is the least stable operating condition, it is proposed that the final flow controller tuning be performed at these conditions near rated reactor pressure. The system can then be retested at lower pressures for verification of control loop settings. After all final controller and system adjustments have been determined, a set of demonstration tests in the RST injection mode are performed with the final settings. These demonstration tests provide the necessary information to demonstrate compliance with Regulatory Guide 1.68 objectives.

CONCLUSION:

Startup Test 14.0, RCIC, requires a final set of demonstration tests to verify system performance. These tests demonstrate compliance with the objectives of Regulatory Guide 1.68, Appendix A, paragraphs 4.k and 5.k. Tuning of the control systems can be done primarily at the most efficient test conditions to minimize unnecessary testing of the system. The proposed change does not adversely affect any safety systems or the safe operation of the plant and therefore does not involve an unreviewed safety question. Startup Test 14.0, RCIC, can therefore be simplified by minimizing tuning of the controllers during low pressure. Controller adjustments will be made during the rated reactor pressure testing of the RST injection mode for the turbine speed controller and the reactor pressure vessel injection mode for the flow controller.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 12:

Identical in principle; the only difference is that tuning of the RCIC controllers at low reactor pressures is minimized at Clinton Power Station and deleted at HCGS.

STARTUP TEST 18.0 - TRAVERSING INCORE PROBE UNCERTAINTY
JUSTIFY TEST DELETION

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2; August 1978), Appendix A, paragraph 5.y requires that the incore neutron flux instrumentation be calibrated as necessary and proper operation verified. The Traversing Incore Probe (TIP) System is one of several incore neutron/gamma flux instrumentation systems. It provides gross core power distribution information for several applications. TIP system operability is demonstrated during preoperational testing and during power ascension testing of the process computer. Startup Test 18.0, Traversing Incore Probe Uncertainty, determines the uncertainty of the TIP system readings. It is proposed to delete Startup Test 18.0.

DISCUSSION:

TIP system operability is demonstrated during preoperational testing of the TIP hardware and electronics and during power ascension testing when the process computer undergoes the dynamic system test case. During the latter testing, the process computer program OD-1 is used in conjunction with the TIP system to provide information on the gross core power distribution. Startup Test 18.0 is a separate test performed later in the power ascension test program. It provides a measure of the uncertainty in TIP system data.

Uncertainty in TIP indication affects the accuracy of Local Power Range Monitor (LPRM) calibrations, thermal limits calculations, operating recommendations, etc. For Startup Test 18.0, the acceptance criterion states that total TIP uncertainty shall be less than 6.0%. With an increased TIP uncertainty, the perceived operating margin will be less than the true value and therefore, if the plant is operating "on limits" with the larger uncertainty, the true operating margin will be greater than if the uncertainties were smaller. Therefore, the larger uncertainties will actually result in larger margins to limits.

Total TIP uncertainty is comprised of geometric and random noise components. Historically, geometric uncertainty resulted from the off-center placement of the TIP tube within the LPRM instrument tube, bowing of the instrument tube, and water gap dimensional variations. These geometric differences cause the thermal neutron TIP detectors to indicate flux levels different from the values ideally obtained by an axial scan down the center of the water gap. However, centering of the TIP tube within the instrument tube for the BWR/6 design removes this contribution to the geometric uncertainty. A measure of this uncertainty is obtained by comparing data from symmetric TIP locations and correcting for random noise uncertainty.

Random noise uncertainty is caused by neutron, electronic and boiling noise in the reactor. This uncertainty is determined by comparing data from repetitive scans in the common instrument tube by each TIP detector.

Measurement of these uncertainties at the beginning-of-life of an initial core, during power ascension testing, provides the best measure of TIP uncertainty caused by these effects because the fuel bundle power asymmetry is at a minimum. Results from previous plant startups show that measured total TIP asymmetry has always been well below the acceptance criterion, 6%. Detailed analysis (Reference 1) of 45 TIP sets from eight plants for power levels ranging from 18% to 100% and core flow from 33% to 105% showed that the average total TIP uncertainty was 3.8%. Results from more recent power ascension testing of 7 plants, summarized in Table 1, show that the average values of the geometric uncertainty, random noise uncertainty and total TIP uncertainty were 1.85, 1.02 and 2.17% respectively. The Kuosheng-1 BWR/6 plant is similar to the Clinton plant and representative of the expected TIP uncertainties at the Clinton Power Station (CPS).

CONCLUSION:

Based on the test results from previous plant startups, TIP uncertainty for CPS is expected to be much less than the limiting value of 6%. TIP system operability will be demonstrated during preoperational testing of the TIP hardware and electronics and during power ascension testing of the process computer. In view of these considerations, it is concluded that deletion of Startup Test 18.0, Traversing Incore Probe Uncertainty, does not adversely affect any safety related systems or the safe operation of the plant and as such does not involve an unreviewed safety question.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 16:

Identical in principle; HCGS replaced the thermal neutron TIP detector with a gamma flux sensitive detector, which reduced the geometric uncertainty of the system.

REFERENCES:

1. B.G. Lord, "Updated and Expanded TIP Asymmetry Analysis for Initial and Reload Cores," General Electric Company, December 2, 1977 (NEDM-11132-131).

Table 1 - TIP Uncertainty Startup Data

	TIP UNCERTAINTY (%)		
	GEOMETRIC	RANDOM	TOTAL
HANFORD-2, TC3	2.87	1.42	3.20
HANFORD-2, TC6	1.80	1.43	2.30
LASALLE-2, TC3	1.24	1.18	1.71
LASALLE-1, TC6	2.16	1.54	2.65
LEIBSTADT, TC3*	2.55	0.68	2.64
LEIBSTADT, TC6*	1.57	0.83	1.78
FUKUSHIMA-6, TC3	1.50	1.00	1.80
FUKUSHIMA-6, TC6	1.30	1.10	1.70
CHINSHAN-1, TC3	2.51	1.21	2.79
CHINSHAN-1, TC6	2.40	0.61	2.48
CHINSHAN-2, TC3	1.20	0.88	1.49
CHINSHAN-2, TC6	0.98	0.59	1.14
CAORSO, TC2 (25% POWER)	1.40	1.14	1.81
CAORSO, TC2 (43% POWER)	1.60	0.98	1.88
CAORSO, TC3 (53% POWER)	1.97	0.95	2.19
CAORSO, TC3 (49% POWER)	1.37	1.10	1.76
CAORSO, TC6 (97% POWER)	2.29	0.73	2.40
CAORSO, TC6 (97% POWER)	2.21	0.94	2.40
KUOSHENG-1, TC3	4.80**	0.78	4.86**
KUOSHENG-1, TC6	2.17	0.86	2.33
SUSQUEHANNA-1, TC3	0.78	1.46	1.66
SUSQUEHANNA-1, TC6	1.08	1.08	1.53
SUSQUEHANNA-1, TC6	0.90	1.08	1.41
MEAN	1.85	1.02	2.17

* Leibstadt has gamma flux-sensitive TIP's.

** TIP axial positioning was incorrectly aligned.

STARTUP TEST 21.0 - CORE POWER-VOID MODE RESPONSE
JUSTIFY TEST DELETION

OBJECTIVE:

There are no specific Regulatory Guide 1.68 requirements to perform stability testing during the power ascension program. However, paragraphs 5.s, 5.v and 5.h.h require the demonstration of acceptable control system responses during steady state and transient conditions. Startup Test 21.0, Core Power-Void Mode Response, measures the stability of the core power-void dynamic response by moving a very high worth control rod one or two notches. In conjunction, Startup Test 22.0, Pressure Regulator, performs pressure regulator step changes to measure the core power-void dynamic response. These tests are currently planned to be performed at Test Conditions 4 and 5. It is proposed to delete the control rod movement tests at Test Conditions 4 and 5 and pressure regulator tests at Test Condition 4, while still maintaining the pressure regulator testing at Test Condition 5.

DISCUSSION:

Acceptable response of the core power-void mode is determined by analyzing test data and comparing to an acceptance criterion which defines the required system performance. The criterion requires that all system-related variables must exhibit non-divergent behavior. System-related variables are heat flux and reactor pressure.

Measurement of system stability by movement of control rods was developed for small reactor cores. Use of this technique for large loosely coupled BWR's, typical of current plants in the startup testing phase, will not provide significant information on the stability of the system because of the low signal-to-noise ratio. In addition, for large BWR cores (i.e., Clinton Power Station), control rod worths in the power/flow range of interest are much less than for a small tightly coupled core. Instead, core wide disturbances provide more meaningful data for large cores. Startup Test 22.0, Pressure Regulator, measures the system reponse to pressure disturbances caused by actions

of the pressure regulator system. This testing yields valuable core stability data at the limiting high power/low flow condition encountered during normal operation (Test Condition 5). In addition to Startup Test 22.0, normal observations of operational power maneuvers provide sufficient data to determine the normal stability characteristics and response of the system.

In addition to the pressure regulator testing, Service Information Letter (SIL) 380 (Reference 1) provides detailed recommendations for the monitoring of system behavior. These recommendations provide for monitoring of neutron flux characteristics during normal operation at high power/low flow conditions and during abnormal operating conditions. In addition to the monitoring requirements, current Technical Specifications do not allow continued operation at natural circulation flow which is the least stable condition of the operating region.

Extensive special testing of stability characteristics has also been performed at several BWR's, including Vermont Yankee, Caorso, Leibstadt, Peach Bottom-2 and Browns Ferry. The test data has demonstrated the stability characteristics of BWR's over a wide range of conditions and has been reviewed along with extensive supporting analyses, as part of the Staff's Safety Evaluation Report on core thermal-hydraulic stability (Reference 2).

CONCLUSION:

As a result of the extensive testing and analysis of core thermal hydraulic stability, it has been demonstrated that General Electric BWR fuel and core designs meet the stability criteria set forth in General Design Criteria 10 and 12 of 10CFR50, Appendix A (Reference 2). Based on the above discussion and the Staff's Safety Evaluation Report (Reference 2), the proposed change will not adversely affect any safety related systems or safe operation of the plant and therefore does

not involve an unreviewed safety question. System stability is adequately measured during Startup Test 22.0, Pressure Regulator, and has been extensively tested at several BWR's covering a wide range of designs. In addition, information on the system's stability is continuously provided by SIL-380 recommendations for the monitoring of neutron flux. Therefore, Startup Test 21.0, Core Power-Void Mode Response, can be deleted from the Power Ascension Test Program.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 19:

This Clinton Power Station test deletion is identical to that submitted for HCGS.

References:

1. "BWR Core Thermal Hydraulic Stability", Service Information Letter 380, Revision 1, General Electric Company, February 10, 1984.

2. Letter, C.O. Thomas (NRC) to H.C. Pfefferlen (GE), "Acceptance For Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, 'Thermal Hydraulic Stability Amendment to GESTAR II'", April 24, 1985.

STARTUP TEST 22.0 - PRESSURE REGULATOR
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2; August 1978), Appendix A, paragraphs 4.u and 5.s require demonstration of the operability during low power testing and calibration and performance verification during power ascension testing of the Pressure Regulator system. Startup Test 22.0, Pressure Regulator, determines the response of the system to rapid pressure setpoint changes and at specified test conditions, the load limit setpoint will be set so that the transient is handled by control and/or bypass valves. Backup pressure regulator takeover will be tested by simulating a failure of the selected pressure regulator. Performance of this test is planned for Test Conditions 1 through 6. It is proposed to delete the Pressure Regulator tests at Test Condition 4 and the backup pressure regulator takeover testing at Test Condition 5.

DISCUSSION:

The pressure regulator system is primarily sensitive to vessel steam flow (and hence, power level) since the reactor is basically operated as a constant pressure device for varying steam flows. Therefore, testing of the pressure regulator response should cover the range of expected core power levels and is not significantly dependent on core flow since the steam flow at a fixed power level is insensitive to the core flow rate. Testing of the pressure regulator system during Test Condition 2, 3, 5 and 6 adequately covers the range of expected power levels during plant operation. Therefore, testing of the pressure regulator system at Test Condition 4 is not required for verification of the controller performance, and testing at Test Conditions 2, 3, 5 and 6 will provide adequate confirmation of the system performance over the entire operating range.

Pressure regulator testing (specifically the pressure setpoint changes) at Test Condition 4 also provides information on the stability of the system. However, information on the stability of the reactor at Test Condition 4 can also be obtained by monitoring the neutron flux (both

local and core average) as required by Technical Specification section 3/4.4.1.1. These surveillance requirements provide for monitoring of the Average Power Range Monitor (APRM) and Local Power Range Monitor (LPRM) detectors when operating at natural circulation conditions and provide sufficient information on the stability of the reactor at Test Condition 4 in addition to providing operator training for the monitoring procedures. Pressure regulator testing (pressure setpoint changes) will be performed at Test Condition 5, which bounds the least stable portion of the normal operating region, and will provide additional information on the stability of the reactor. Therefore, pressure regulator testing at Test Condition 4 can be deleted.

Testing of the backup pressure regulator is performed by simulating the failure of a selected pressure regulator. This test is currently planned to be performed at Test Conditions 2, 3, 5 and 6. Testing at Test Conditions 2, 3 and 6 provides adequate demonstration of the capability of the backup pressure regulator to control pressure in the event of a failure of the controlling pressure regulator since these test conditions bound the power level of Test Condition 5.

CONCLUSION:

Testing of the Pressure Regulator system at Test Condition 4 is not required since the pressure regulator will be tested at other Test Conditions that bound the power level of Test Condition 4 and stability data during Test Condition 4 will be obtained by monitoring of the APRM and LPRM detectors as required by Technical Specifications. Therefore, deletion of Test Condition 4 testing does not adversely affect any safety related systems or the safe operation of the plant and does not involve an unreviewed safety question. Testing of the backup pressure regulator at Test Conditions 2, 3 and 6 demonstrates the performance of the backup system. Deleting testing of the backup pressure regulator at Test Condition 5 does not adversely affect any safety systems or the safe operation of the plant and therefore does not involve an unreviewed safety question. Startup Test 22.0, Pressure Regulator, can therefore be simplified by deleting the backup pressure regulator testing at Test Condition 5 and the testing at Test Condition 4.

STARTUP TEST 22.0 - PRESSURE REGULATOR
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 20:

Identical in principle; with the exception that the automatic load following feature testing will not be deleted at Clinton Power Station.

STARTUP TEST 23.1 - FEEDWATER SYSTEM RESPONSE
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 4.u, 5.s and 5.v require the demonstration of operability during low power testing, the calibration and performance verification during power ascension testing and the verification of operation in accordance with design requirements during power ascension testing of the feedwater control system. Startup Test 23.1, Feedwater System Response, verifies that the feedwater system has been adjusted to provide acceptable reactor water level control. Nominal water level setpoint changes are used to evaluate the feedwater control system settings for all power and feedwater pump modes. Performance of this test is currently planned for Test Conditions 2 through 6. It is proposed to delete water level setpoint and manual feedwater flow tests at Test Condition 4.

DISCUSSION:

Acceptance criteria define acceptable performance of the feedwater control system to testing perturbations. Criteria require that the transient response of any level control system-related variable to any test input must be non-divergent and also specify response characteristics for given disturbances. Testing during the power ascension program is performed at Test Conditions 2 through 6 to demonstrate compliance to these criteria.

The feedwater control system maintains the mass balance of the reactor vessel by supplying water to the vessel to match the steam flow exiting the vessel, thereby maintaining a constant water level during normal operation. Therefore, the feedwater control system is primarily dependent on the vessel steam flow and hence the reactor power. Testing of the feedwater control system at Test Conditions 2, 3, 5 and 6 adequately bounds the expected power levels for system operation. Since the power level of Test Condition 4 is similar to that of Test

Condition 5, the feedwater control system performance at Test Condition 4 is not expected to be significantly different than at Test Condition 5. Therefore, testing at Test Conditions 2, 3, 5 and 6 will adequately confirm the system performance over the entire operating range. The objectives of Regulatory Guide 1.68, paragraphs 4.u, 5.s and 5.v are still satisfied with the remaining testing.

CONCLUSION:

Testing of the feedwater control system at Test Conditions 2, 3, 5 and 6 provides demonstration of system performance over the entire operating range. As such, deletion of Test Condition 4 testing does not adversely affect any safety systems or the safe operation of the plant and as such does not involve an unreviewed safety question. The proposed testing satisfies the objectives of Regulatory Guide 1.68, Appendix A, paragraphs 4.u, 5.s and 5.v, as well as the requirements of Startup Test 23.1. Therefore, Startup Test 23.1, Feedwater System Response, can be simplified by deleting testing at Test Condition 4.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 21A:

This Clinton Power Station test simplification is identical to that submitted for HCGS.

STARTUP TEST 24.0 - TURBINE VALVE SURVEILLANCE
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 4.q and 5.t require demonstration of the operability of the turbine stop valves, turbine control valves and turbine bypass valves. Startup Test 24.0, Turbine Valve Surveillance, determines the operability of the turbine stop valves (TSV), turbine control valves (TCV) and turbine bypass valves (BPV) and determines the maximum power level at which surveillance can be performed with ample margin to scram. It is proposed to define a maximum power for TSV, TCV and BPV surveillance testing based on previous BWR test experience rather than determining the maximum power during the power ascension test program.

DISCUSSION:

Individual main turbine control, stop and bypass valves are tested routinely during plant operation as required for turbine surveillance testing. At several test points, the response of the reactor is determined by analyzing test data and comparing the results to acceptance criteria which define the limiting system performance. The criteria require that peak values for system variables (neutron flux, heat flux, vessel pressure and steam flow) do not exceed prescribed limits.

Startup testing of the turbine valves is initially performed between Test Conditions 2 and 3 at 45-65% power, which satisfies the objectives of Regulatory Guide 1.68. This initial functional test is also used to extrapolate to another test point and finally to a maximum power test condition with ample margin to scram. The proposed test change deletes determining the maximum power condition for turbine valve surveillance testing, while retaining the initial functional testing between Test Conditions 2 and 3. The maximum power condition for all subsequent testing will be selected by evaluating the data base from previous BWR

startups and operation. The adequacy of this power level will be demonstrated by the turbine valve surveillance testing required by the Technical Specification surveillance requirements. Testing to determine the maximum power level at which the turbine valve surveillance testing can be performed with ample margin to scram can be performed at a later time if desired.

CONCLUSION:

Functional testing of the turbine valves is performed between Test Conditions 2 and 3. Additional testing to determine the maximum power level for future surveillance testing is not required and therefore the proposed testing will satisfy all objectives of Regulatory Guide 1.68, Appendix A, paragraph 4.q and 5.t. The proposed change will not adversely affect any safety systems or the safe operation of the plant, and as such, does not involve an unreviewed safety question.

SIMILARITY WITH HOPE CREEK GENERATING STATION TEST NUMBER 22:

This test simplification is unique to Clinton Power Station.

STARTUP TEST 25.1 - MAIN STEAM ISOLATION VALVE FUNCTIONAL TESTS
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 4.1 and 5.u require demonstration of the operability of the main steam isolation valves (MSIV). These paragraphs also require the demonstration of stroke times at rated temperature and pressure conditions and the response times during power ascension testing for the MSIV's. Startup Test 25.1, Main Steam Isolation Valve Functional Tests, determines the functional performance and individual fast closure time of each MSIV at heatup and between Test Conditions 1 and 3. The maximum power level at which full individual closure can be performed without scram is also determined during Startup Test 25.1 by closing the fastest MSIV at Test Condition 2 or 3 then again between Test Conditions 5 and 6, the latter test point being determined from extrapolation of the first test data. Further extrapolation of the higher power test data determines the maximum power where all subsequent surveillance testing will be performed. It is proposed to define a maximum power level for future MSIV surveillance testing based on previous BWR experience rather than determining the maximum power during the power ascension test program.

DISCUSSION:

The response of the reactor and its associated systems is determined during the tests to find the maximum power level for future surveillance testing by analyzing test data and comparing the results to acceptance criteria, level 1 and level 2, which define the required system performance. Level 1 criteria require that closure and delay times for each MSIV be within specified limits. The level 2 criteria require that the reactor shall not scram or isolate on individual MSIV closures, and that peak values for system variables (neutron flux, heat flux, vessel pressure and steam flow) do not exceed prescribed limits.

STARTUP TEST 25.1 - MAIN STEAM ISOLATION VALVE FUNCTIONAL TESTS
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

During Test Condition Heatup and between Test Conditions 1 and 3, individual fast closure of each MSIV will be performed to verify their functional performance and to determine closure times. This testing satisfies the requirements of Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 4.1 and 5.u. Additional testing of the MSIV's to determine the maximum power condition for future MSIV functional testing is not required by Regulatory Guide 1.68 and can be deleted from the power ascension test program. The maximum power condition for all subsequent testing will be selected by evaluating the data base from previous BWR startups and operation. The adequacy of this power level will be demonstrated by the MSIV testing required by Plant Technical Specification surveillance requirements. Testing to determine the maximum power level at which the MSIV surveillance testing can be performed with ample margin to scram can be performed at a later time if desired.

CONCLUSION:

Testing of the MSIV's is performed during Test Condition Heatup and between Test Conditions 1 and 3. Additional testing to determine the maximum power level for future surveillance testing is not required and therefore the proposed testing will satisfy the objectives of Regulatory Guide 1.68, Appendix A, paragraph 4.1 and 5.u, as well as the requirements of Startup Test 25.1. The proposed change will not adversely affect any safety systems or the safe operation of the plant, and therefore, does not involve an unreviewed safety question.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 23A:

This CPS test simplification is identical to that submitted for HCGS.

STARTUP TEST 26.0 - RELIEF VALVES
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 4.p requires testing of the main steam relief valves to demonstrate operability of the valves at rated temperature during low power testing. Paragraph 5.t requires that if not previously accomplished, the licensee should verify, if appropriate, the operability, response times, relieving capacities, setpoints and reset pressures for the main steam line relief valves during power ascension testing. Startup Test 26.0, Relief Valves, is performed to demonstrate operability of the relief valves at rated temperature and verify that major blockages that could affect valve capacity do not exist. Bench tests and preoperational tests verify other valve characteristics. It is proposed to reduce the relief valve tests for Startup Test 26.0 to those required to assure plant safety so that the potential for test induced damage to the valves is minimized. This will be accomplished by deleting the tests during heatup at 250 psig and replacing the tests performed between Test Conditions 2 and 3 with new tests at Test Condition 1.

DISCUSSION:

ASME capacity certification test data and dimensional measurements of valve throat area assure that valve capacity meets the design requirements. Bench testing previously performed on each valve with steam at valve normal operating conditions verify that response times, setpoints and reset pressures satisfy the design requirements. Preoperational tests confirm that pressure signal, power supply and air supply logic are as intended.

Acceptable operation of the relief valves is determined by analyzing test data and comparing to acceptance criteria which define the required performance. The criteria require that there is positive indication of steam discharge during a manual actuation of each valve, that the pressure control system response is stable, and that the discharge temperature remains within acceptable limits. These criteria can be demonstrated during single valve testing at rated reactor pressure with steam flow greater than the relief valve capacity.

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 4.p requires demonstration of relief valve operability at rated temperature during low power testing (defined as "normally at less than 5% power"). To provide adequate control of system pressure, the testing must be performed at a steam flow that is greater than the individual relief valve capacity. Since the relief valve capacity is typically 5-7% steam flow, testing is proposed to be done at approximately 10-20% power (Test Condition 1), approximately the test power level of paragraph 5.t of Regulatory Guide 1.68.

Actuation of relief valves at low pressure has been identified as a contributor to valve seat damage caused by reseating against abnormally low pressure. Therefore, relief valve testing at low pressure should be minimized to reduce unnecessary damage to the valve seats. Previous testing at 250 psig was performed to ensure that relief valves would function properly at rated pressure if depressurization was required. This testing responded to occurrences where high air supply pressure or incorrectly wired logic circuits resulted in failure of energized solenoids to de-energize and resultant valve opening. In response to these instances, qualification tests performed on these valve designs have demonstrated that the design satisfies specified requirements. In addition, bench tests are performed on each relief valve to provide assurance that each assembly will perform satisfactorily and preoperational tests check out the adequacy of electrical power supply, logic and air supply.

STARTUP TEST 26.0 - RELIEF VALVES
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

Requirements to test the Automatic Depressurization System (ADS) valves as required by Plant Technical Specification surveillance requirements to ensure that depressurization capability exists are still maintained with the proposed testing . Overpressure protection during transients at power levels below 25% power can be adequately handled by the safety valves. Therefore, single valve testing at Test Condition 1 meets the objective of demonstrating relief valve operability required by Regulatory Guide 1.68, Appendix A, paragraphs 4.p and 5.t.

CONCLUSION:

Testing of the relief valves at Test Condition 1 demonstrates the operability of the relief valves. This proposed testing change does not adversely affect any safety systems or safe operation of the plant and therefore does not involve an unreviewed safety question. Startup Test 26.0, Relief Valves, can therefore be simplified by replacing the testing at low pressure (250 psig) and between Test Condition 2 and 3 with testing at Test Condition 1 coincident with steam flow greater than the relief valve capacity.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 24:

This Clinton Power Station test simplification is identical to that submitted for HCGS.

STARTUP TEST 27.0 - TURBINE TRIP AND GENERATOR LOAD REJECTION
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 5.1.1 and 5.n.n require that a Turbine Trip and Generator Load Rejection be performed at 100% power to demonstrate that the dynamic response of the plant is in accordance with design requirements for turbine trip and full load rejection. These tests may be combined if a turbine trip is initiated directly during the generator load rejection instead of tripping from secondary effects such as a turbine overspeed trip. Startup Test 27.0, Turbine Trip and Generator Load Rejection, is currently planned to be performed at three conditions during the power ascension test program: (1) a generator load rejection during Test Condition 2 (within the bypass capacity of the plant); (2) a turbine trip during Test Condition 3 (approximately 75% power); and (3) a generator load rejection at Test Condition 6 (approximately 100% power). It is proposed to delete the turbine trip test at Test Condition 3 and change the generator load rejection test at Test Condition 2 to a turbine trip test. This proposed testing will demonstrate that Regulatory Guide 1.68 objectives are met.

DISCUSSION:

Response of the system during a turbine trip and generator load rejection is determined by analyzing test data and comparing to acceptance criteria, level 1 and level 2, which define the required system performance. Level 1 criteria require proper operation of the turbine control and stop valve closure times with respect to the bypass valve opening time, adequate bypass valve response times, proper feedwater control system level response to prevent flooding of the steam lines, that recirculation flow coastdown following protective trips is within design values, acceptable vessel dome pressure and simulated heat flux response, and proper operation of the low-low set pressure relief logic for the safety/relief valves.

Level 2 criteria require that no MSIV closure occur during the first three minutes of the event, that vessel dome pressure and simulated heat flux changes do not exceed predicted values, that low water level recirculation pump trip is avoided, that High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) Systems are not initiated, that feedwater level control avoids loss of feedwater because of high level trips, and that safety/relief valve discharge temperatures remain within acceptable limits. In addition, for the generator load rejection within bypass capacity (Test Condition 2), level 2 criteria require that the reactor does not scram and that the bypass capacity calculated is greater than or equal to the assumed value in FSAR analysis.

The generator load rejection (Test Condition 6) and the turbine trip (Test Condition 2) will provide data to demonstrate that the level 1 and 2 criteria are met during a turbine trip. The turbine bypass system performance will be verified at a lower power level by changing the proposed generator load rejection at Test Condition 2 to a turbine trip. Integrated system response to a turbine trip can be obtained from the generator load rejection test at Test Condition 6.

Control systems which regulate the long term operation following the transients are separately tested during the power ascension test program. Feedwater and level control system tuning in Startup Test 23.1, Feedwater System Response, will ensure proper water level control. High and low water level trip avoidance will be verified in the generator load rejection test at Test Condition 6.

CONCLUSIONS:

The turbine trip test has been previously demonstrated to be a mild transient event and poses no serious threat to the core and reactor integrity. In addition, the transient results from a generator load rejection at full power are more limiting than the results from a turbine trip at Test Conditions 3 and 6. Based on the above discussions, the proposed change will not adversely affect the safety related systems or safe operation of the plant and therefore does not involve an unreviewed safety question.

Current testing of the generator load rejection at 100% power satisfies the requirements imposed by Regulatory Guide 1.68 (Revision 2), Appendix A, paragraphs 5.1.1 and 5.n.n. In addition, the proposed turbine trip test within bypass valve capacity (Test Condition 2) provides additional verification of the response of the protective systems and also provides demonstration of the bypass system's capability to avoid scram at low power levels. Therefore, the turbine trip at Test Condition 3 can be deleted with the added change that the generator load rejection at Test Condition 2 will be changed to a turbine trip test.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 25:

This Clinton Power Station test simplification is identical to that submitted for HCGS.

STARTUP TEST 29.0 - RECIRCULATION FLOW CONTROL SYSTEM
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 5.s requires that the recirculation flow control system be calibrated as necessary and performance verified, and paragraph 5.h.h requires that the dynamic response of the plant to design load swings be demonstrated to be in accordance with design. Startup Test 29.0, Recirculation Flow Control System, determines the plant response to changes in recirculation flow and optimizes settings of the recirculation flow controller. Testing is performed over a wide range of power/flow conditions. It is proposed to simplify the testing by reducing the number of intermediate flow conditions and testing inputs (ramp and step demands).

DISCUSSION:

Response of the system is determined by analyzing test data and comparing to acceptance criteria which define the required system performance. The testing of the Recirculation Flow Control System follows a "building block" approach while the plant is ascending from low to high power levels. Components and inner control loops are tested first, followed by drive flow control and plant power maneuvers to adjust and then demonstrate the outer loop controller performance. Testing is performed to demonstrate the capability of the system over the entire flow control range.

Prior to the power ascension test program, predictions of system behavior are performed to aid in the tuning of the Recirculation Flow Control System. In addition, benchmark testing of the flow control valve recirculation system and successful startup test results from several flow control valve plants (Kuosheng 1 and 2, LaSalle 1 and 2 and Leibstadt) provide valuable experience which can be used to reduce

the number of test cases while still fulfilling the purpose of Startup Test 29.0. As a result, the number of intermediate flow conditions and test inputs (ramp and step demands) required to achieve the desired system performance can be reduced.

CONCLUSIONS:

By using pretest analysis and experience from previous flow control valve plant startup testing, the objectives of Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraphs 5.s and 5.h.h can be satisfied with the proposed simplified testing. The proposed changes will not adversely affect any safety related systems or the safe operation of the plant and therefore does not involve an unreviewed safety question. Therefore, Startup Test 29.0, Recirculation Flow Control System, can be simplified by reducing the number of intermediate flow conditions and test inputs (ramp and step demands).

SIMILARITY WITH HOPE CREEK GENERATING STATION TEST NUMBER 27:

Identical in principle; the differences are (1) the automatic load following feature will be tested at Clinton Power Station and (2) Illinois Power Company will rely on the experience base from tuning tests performed at other similar BWR plants.

STARTUP TEST 30.1 - SINGLE RECIRCULATION PUMP TRIP
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 5.i.i requires that the dynamic response of the plant is in accordance with design for limiting reactor coolant pump trips. The method for initiating the pump trip should result in the fastest credible coastdown in flow for the system (recommended at 100% power). Startup Test 30.1, Single Recirculation Pump Trip, is performed at Test Conditions 3 and 6. The tests are performed to obtain recirculation system performance during the pump trip and pump restart and to verify feedwater control system capability to control water level without resulting in a high water level scram. It is proposed to delete the one pump trip at Test Condition 6.

DISCUSSION:

Response of the system during a single recirculation pump trip is determined by analyzing test data and comparing to acceptance criteria which define the required system performance. For the single recirculation pump trip test, the reactor water level, simulated heat flux and Average Power Range Monitor (APRM) signal must have margin to scram setpoints. These criteria are applied to provide assurance that the system can avoid a scram during a single pump trip to improve plant availability. The testing is not required to verify Final Safety Analysis Report (FSAR) transient analysis since it has been demonstrated by previous BWR/6 startup testing and analytically (FSAR) that the single pump trip has a negligible impact on safety limits. In addition, the characteristics of recirculation pump trips are well understood and have been demonstrated during power ascension testing at previous plants.

Testing of a single pump trip at Test Condition 3 will be performed to demonstrate the system performance under these conditions. Based on the data from the two pump trip, the system performance during a trip of the other recirculation pump can then be estimated.

Startup Test 16.0, Selected Process Temperatures, requires vessel and recirculation loop temperatures to be measured during planned recirculation pump trips to verify that temperature stratification does not occur in the idle recirculation loops or in the lower plenum when one or more loops are inactive. This temperature data is also required by Technical Specification section 3/4.4.1 which limits the temperature differential between the recirculation loops and the vessel. Therefore, compliance with this Technical Specification, when applicable, will satisfy the requirements of Startup Test 16.0 following a recirculation pump(s) trip.

Startup Test 33.2, Drywell Piping Vibration, verifies that recirculation piping vibration is within acceptable limits. Remote vibration and deflection measurements are taken during recirculation pump starts and pump trips at approximately 100% of rated flow. Previous plant startup results from Kuosheng-1 and 2 (Clinton prototype) indicate that vibration and deflection measurements of recirculation piping during recirculation pump trips and restarts are always well within the expected range and are significantly below the allowable range (Table 1). In addition, recirculation piping vibration data will be recorded during the single recirculation pump trip test at Test Condition 3 (100% core flow). Recirculation pump trips from 100% core flow provide the limiting conditions for recirculation piping vibration testing because of the faster coastdown of the pumps at higher flow. These results are indicative of the expected response of the Clinton Power Station (CPS) and, since the recirculation piping vibration data will be obtained at Test Condition 3, provide the basis for deleting the recirculation pump trip and restart vibration data during the single pump trip at Test Condition 6.

CONCLUSION:

Performance of single pump trips are only required as demonstration of the ability of the feedwater control system to improve plant availability by avoiding a high water level scram during the single pump trip.

Therefore, deleting the single pump trip at Test Condition 6 will not adversely affect any safety related systems or safe operation of the plant and therefore does not involve an unreviewed safety question. Data for Startup Test 16.0, Selected Process Temperatures, can be obtained during the single pump trip from Test Condition 3. In addition, previous plant startup results have demonstrated that the vibration and deflection measurements during recirculation pump trips and restarts for plants similar to CPS are well below prescribed limits and the piping vibration data will be recorded at Test Condition 3. Therefore, this vibration data is not required during the single pump trip at Test Condition 6. Therefore, Startup Test 30.1, Single Recirculation Pump Trip, can be deleted at Test Condition 6.

SIMILARITY WITH HOPE CREEK GENERATING STATION TEST NUMBER 20A:

This test simplification is unique to CPS.

TABLE 1

KUOSHENG-1,2 TRANSIENT VIBRATION TESTS

EVENT	TEST CONDITION	AS MEASURED MAXIMUM PEAK-TO-PEAK (MILS)	LEVEL 1 ALLOWABLE RANGE (MILS)	LEVEL 2 EXPECTED RANGE (MILS)
Single Pump Trip	6	36	150	68
Two Pump Trip	6	43	150	68
Two Pump Trip	3	19	275	100

STARTUP TEST 30.3 - RECIRCULATION PUMP TRIP - TWO PUMPS
IN CONJUNCTION WITH STARTUP TEST 27.0

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 5.i.i requires that the dynamic response of the plant is in accordance with design for limiting reactor coolant pump trips. The method for initiating the pump trip should result in the fastest credible coastdown in flow for the system (recommended at 100% power). Startup Test 30.3, Recirculation Pump Trip - Two Pumps, is currently planned to be performed at Test Condition 3 as a preliminary check of the Recirculation Pump Trip (RPT) circuit and the recirculation pump flow coastdown. The two pump trip will be initiated by simultaneously tripping both RPT breakers using a test switch. It is proposed to perform this test in conjunction with Startup Test 27.0, Generator Load Rejection, at Test Condition 6. This testing will demonstrate that Regulatory Guide 1.68, Appendix A, paragraph 5.i.i objectives are met for a two pump trip.

DISCUSSION:

Response of the system during a two pump trip is determined by analyzing test data and comparing to acceptance criteria which define the required system performance. For the two pump trip test, the recirculation drive flow coastdown must be within specified limits used in the Final Safety Analysis Report (FSAR) transient analyses. During Startup Test 27.0, Turbine Trip and Generator Load Rejection at Test Condition 6, a two pump trip occurs as the result of the protective RPT function. This testing results in an actual demonstration of the RPT circuit. The flow coastdown data, which is negligibly affected by the load rejection transient, is recorded during the transient to demonstrate compliance with Regulatory Guide 1.68 requirements. Testing the recirculation drive flow coastdown at Test Condition 3 will not satisfy the requirements of Regulatory Guide 1.68 since the pump coastdown is more limiting at higher power levels. The load rejection is performed at Test Condition 6 (approximately 100% power) as required by paragraph 5.i.i of Regulatory Guide 1.68.

STARTUP TEST 30.3 - RECIRCULATION PUMP TRIP - TWO PUMPS
IN CONJUNCTION WITH STARTUP TEST 27.0

The characteristics of recirculation pump trips are well understood and have been demonstrated during power ascension at previous plants. The two pump RPT event has a negligible effect on plant parameters as demonstrated in the FSAR and therefore the two pump trip transient test is not required to verify FSAR transient analyses.

Operability of the RPT function prior to its demonstration during Startup Test 27.0 at Test Condition 6 is demonstrated during preoperational testing. In addition, Technical Specification section 3/4.3.4.2, End-of-Cycle RPT System Instrumentation, requires that the instrumentation channel be demonstrated operable at least once every 31 days. These tests provide adequate assurance that the RPT will operate properly during Startup Test 27.0 at Test Condition 6.

Startup Test 33.2, Drywell Piping Vibration, verifies that recirculation piping vibration is within acceptable limits. Remote vibration and deflection measurements are taken during recirculation pump starts and pump trips at approximately 100% of rated flow. Previous plant startup results at Kuosheng 1 and 2 (Clinton prototype) indicate that vibration and deflection measurements of recirculation piping during recirculation pump trips and restarts are always well within the expected range and are significantly below the allowable range (Table 1). In addition, recirculation piping vibration data will be recorded during the two recirculation pump trip performed in conjunction with Startup Test 27.0, Turbine Trip and Generator Load Rejection at Test Condition 6 (100% core flow). Recirculation pump trips from 100% core flow provide the limiting conditions for recirculation piping vibration testing because of the faster coastdown of the pumps at higher flow. These results are indicative of the expected response of Clinton Power Station (CPS) and, along with planned recording of the recirculation piping vibration data during the generator load rejection test, provide the basis for deleting the recirculation pump trip and restart vibration data during the two pump trip at Test Condition 3.

STARTUP TEST 30.3 - RECIRCULATION PUMP TRIP - TWO PUMPS
IN CONJUNCTION WITH STARTUP TEST 27.0

CONCLUSION:

Testing of the two pump RPT function, in conjunction with the generator load rejection at Test Condition 6, demonstrates the actuation of the RPT circuits and verifies the drive flow coastdown characteristics. This testing satisfies the objectives of Regulatory Guide 1.68, Appendix A, paragraph 5.i.i and will not adversely affect any safety related systems or safe operation of the plant and therefore does not involve an unreviewed safety question. In addition, previous plant startup results for plants similar to CPS have demonstrated that the vibration and deflection measurements during recirculation pump trips and restarts are well below prescribed limits and the piping vibration data will be recorded at Test Condition 6 (100% core flow) in conjunction with Startup Test 27.0, Turbine Trip and Generator Load Rejection. Therefore, this data is not required during the two pump trip. Therefore, Startup Test 30.3, RPT Trip - Two Pumps, can be performed in conjunction with Startup Test 27.0, Turbine Trip and Generator Load Rejection at Test Condition 6.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 28B:

This CPS test simplification is identical to that submitted for HCGS.

TABLE 1

KUOSHENG-1,2 TRANSIENT VIBRATION TESTS

EVENT	TEST CONDITION	AS MEASURED MAXIMUM PEAK-TO-PEAK (MILS)	LEVEL 1 ALLOWABLE RANGE (MILS)	LEVEL 2 EXPECTED RANGE (MILS)
Single Pump Trip	6	36	150	68
Two Pump Trip	6	43	150	68
Two Pump Trip	3	19	275	100

STARTUP TEST 30.4 - RECIRCULATION RUNBACK IN CONJUNCTION WITH STARTUP TEST 23.3

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 5.s requires that the recirculation flow control system be calibrated as necessary and its performance verified. One function of the recirculation flow control system is to provide a recirculation flow runback upon the coincident loss of one feedwater pump and low water level (Level 4) indication to avoid scram on low-low water level (Level 3). Startup Test 30.4, Recirculation Runback, simulates a loss of feedwater pump at Test Condition 3 near rated recirculation flow to determine the adequacy of the recirculation flow runback feature in preventing a scram. It is proposed to demonstrate this feature in conjunction with Startup Test 23.3, Feedwater Pump Trip at Test Condition 6.

DISCUSSION:

Response of the system during a feedwater pump trip with recirculation runback is determined by analyzing test data and comparing to acceptance criteria which define the required system performance. For the recirculation flow runback test, the recirculation flow control valves are required to runback upon a trip of the runback circuit. Although the purpose of Startup Test 30.4 is to determine the adequacy of the recirculation flow runback feature to prevent a scram following a feedwater pump trip, this is not demonstrated since an actual feedwater pump trip is not initiated. During Startup Test 23.3, Feedwater Pump Trip at Test Condition 6, a recirculation flow runback occurs as the result of the feedwater pump trip. This testing results in an actual demonstration of the recirculation runback circuit under real as opposed to simulated conditions. In addition, the recirculation flow runback test is not required to verify Final Safety Analysis Report (FSAR) analysis results since the feedwater pump trip analysis does not take credit for the automatic runback feature. Therefore, this testing will demonstrate that Regulatory Guide 1.68, Appendix A, paragraph 5.s objectives are met for the recirculation flow runback feature of the recirculation flow control system.

CONCLUSION:

Testing of the recirculation flow runback feature in conjunction with the Feedwater Pump Trip test at Test Condition 6 demonstrates the actuation of the recirculation runback circuits and provides demonstration of the adequacy of the runback feature to prevent scram. This proposed testing satisfies the objectives of Regulatory Guide 1.68, Appendix A, paragraph 5.s for the recirculation flow runback feature of the recirculation flow control system and will not adversely affect any safety related systems or safe operation of the plant and therefore does not involve an unreviewed safety question. Therefore, the performance of the recirculation flow runback feature can be demonstrated in conjunction with Startup Test 23.3, Feedwater Pump Trip at Test Condition 6.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 28D:

This Clinton Power Station test simplification is identical to that submitted by HCGS.

STARTUP TEST 30.5 - RECIRCULATION SYSTEM CAVITATION TEST SIMPLIFICATION

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 5.s requires that the recirculation flow control system be calibrated as necessary and its performance verified. At conditions of high flow and low power, both the jet pumps and the recirculation pumps may cavitate. The analytically-determined cavitation region of the power-flow map is protected by cavitation interlocks which will run back recirculation flow at low power if the cavitation limits are exceeded. The recirculation system flow control valves (FCV) will cavitate at conditions of low power and low subcooling. Cavitation interlocks are also provided to protect against FCV cavitation and will run back recirculation flow upon sensing a decrease in subcooling (as measured by a low feedwater flow). In both of the above cases, flow runback is caused by a shift in the power supply to the recirculation pump motors from normal power to the low frequency motor generators.

Startup Test 30.5, Recirculation System Cavitation, verifies that no recirculation system cavitation will occur in the operable region of the power-flow map. Currently, this test is planned to be performed at Test Conditions 2 and 3, where power will be lowered until a recirculation flow runback occurs, and verifying that cavitation has not occurred. It is proposed that the testing be simplified such that operation at the analytically-determined cavitation limit is performed to determine if cavitation occurs and if the recirculation runback feature is actuated (however, the interlock will be temporarily bypassed to prevent runback of the recirculation flow).

DISCUSSION:

Response of the system near the cavitation region is determined by analyzing test data and comparing to acceptance criteria which define the required system performance. For the recirculation system cavitation test, the recirculation runback logic is required to have settings adequate to prevent operation in areas of potential cavitation.

This may be accomplished without requiring that recirculation runback occur. The recirculation flow runback logic can be bypassed temporarily during Startup Test 30.5. Power can then be reduced by inserting control rods down to the cavitation interlock and verifying that the interlock is actuated (although actual recirculation flow runback will not occur). Signals used to detect pump or FCV cavitation can then be monitored to verify that cavitation has not occurred. With appropriate placement of the jumper on the cavitation interlock, no other recirculation flow runback logic feature will be affected. Should a feedwater transient occur during the performance of this test, the operators can manually runback recirculation flow as necessary to prevent cavitation. The cavitation interlock is not a safety related feature and therefore does not adversely affect the safe operation of the plant.

CONCLUSION:

Cavitation interlocks designed to prevent operation in regions of potential pump or FCV cavitation are analytically determined on a conservative basis. The proposed simplified testing will verify that cavitation does not occur at or above the cavitation interlocks. This proposed testing demonstrates that the acceptance criteria are satisfied and will not adversely affect any safety related systems or safe operation of the plant and does not involve an unreviewed safety question. Therefore, Startup Test 30.5, Recirculation System Cavitation, can be simplified as stated above.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST NUMBER 28E:

Identical in principle; except that Clinton Power Station is performing this startup test at Test Conditions 2 and 3 rather than just at Test Condition 3 as at HCGS.

TEST CONDITION 4 - NATURAL CIRCULATION OPERATION
TEST SIMPLIFICATION - REDUCED NUMBER OF TESTS

OBJECTIVE:

Regulatory Guide 1.68 (Revision 2, August 1978), Appendix A, paragraph 5 requires that "appropriate consideration should be given to testing at the extremes of possible operating modes for facility systems. Testing under simulated conditions of maximum and minimum equipment availability within systems should be accomplished if the facility is intended to be operated in these modes." Test Condition 4 defines the region of the power/flow map on the natural circulation core flow line within ± 5% of the intersection of the 100% rod line. Testing of core performance and control system response is currently planned to be performed in this region. It is proposed to simplify testing in Test Condition 4 by reducing the number of tests.

DISCUSSION:

Although operation at natural circulation conditions is not an intended mode of operation for a BWR, the condition can be reached as the result of a moderate frequency event (two recirculation pump trip, 0.25 events/plant-year). Section 3/4.4.1.1 of the Technical Specifications requires that with no reactor coolant system recirculation loops in operation, the operator should immediately initiate actions to reduce thermal power to less than or equal to the limit specified in Figure 3.4.1.1-1 of the Technical Specifications (approximately the 80% rod line) within four hours. In addition, the operator must initiate measures to place the unit in at least the startup mode within six hours and in hot shutdown mode within the next six hours. Therefore, testing is required at natural circulation conditions to verify operator procedures and to determine system performance.

Currently, testing at Test Condition 4 includes tests to determine control systems response (pressure regulator and feedwater control system), core performance, selected process temperatures, recirculation

system performance and core stability (control rod and pressure perturbations). The recirculation system performance also includes restarting of the recirculation pumps. It is proposed to delete the control systems tuning/demonstration tests during Test Condition 4. The control systems are tested at all other test conditions and their performance is adequately demonstrated over the power/flow range expected during normal operation. Therefore, Startup Test 22.0, Pressure Regulator, and Startup Test 23.1, Feedwater System Response, would be deleted from testing at Test Condition 4.

It is also proposed to replace the stability testing performed at Test Condition 4 (Startup Test 21.0, Core Power-Void Mode Response, and portions of Startup Test 22.0, Pressure Regulator) with the surveillance requirements of Technical Specification section 3/4.4.1.1. This Technical Specification section requires monitoring of the Average Power Range Monitor (APRM) and Local Power Range Monitor (LPRM) detectors when operating at natural circulation conditions. A set of baseline data is required as a measure of the system characteristics, and will be used to determine acceptable plant behavior during later operation. These surveillance requirements have been approved for use by the NRC staff (Reference 1) and are based on recommendations provided by General Electric to all utilities (Reference 2). Stability testing of the reactor will be demonstrated in the normal operating range by the testing performed in conjunction with Startup Test 22.0, Pressure Regulator, at Test Condition 5. In addition to these plant specific tests, significant special testing has been performed at other BWR plants which characterizes the stability performance for BWR's.

The proposed simplified Test Condition 4 testing would therefore include the following: (1) trip of the recirculation pumps near the rated rod line from Test Condition 5, (2) insertion of control rods to approximately the 80% rod line in accordance with section 3/4.4.1.1 of the Technical Specifications, (3) monitoring of steady state recirculation system performance (Startup Test 30.2, Recirculation System), (4) monitoring of APRM and LPRM detectors (Technical Specification section 3/4.4.1.1), (5) monitoring of selected process

temperatures (Startup Test 16.0, Selected Process Temperatures), (6) monitoring of core performance (Startup Test 19.0, Core Performance), and (7) restarting of the recirculation pumps. This proposed simplified testing will provide for operator training during natural circulation operation and will adequately characterize the system performance during these conditions.

CONCLUSION:

Testing of the plant systems is performed over a wide range of operating conditions representing the extremes of possible operating modes intended for the plant. In addition, testing is performed at natural circulation, which although it is not an intended mode of operation, can be encountered following a recirculation pump trip event. The proposed simplified testing provides for adequate operator training and demonstration of system characteristics during Test Condition 4 and therefore meets the objectives of Regulatory Guide 1.68, Appendix A, paragraph 5. The proposed test change does not adversely affect any safety system or the safe operation of the plant and therefore does not involve an unreviewed safety question. Therefore, testing in Test Condition 4 can be simplified as stated above.

SIMILARITY WITH HOPE CREEK GENERATING STATION (HCGS) TEST CONDITION 4 TESTING:

The proposed change at Clinton Power Station is to reduce the number of tests at Test Condition 4 rather than deleting all testing as proposed for HCGS.

REFERENCES:

1. Letter, C.O. Thomas (NRC) to H.C. Pfefferlen (GE), "Acceptance For Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, 'Thermal Hydraulic Stability Amendment to GESTAR II'", April 24, 1985.

2. "BWR Core Thermal Hydraulic Stability," Service Information Letter 380, Revision 1, General Electric Company, February 10, 1984.