Arizona Public Service Company

Director of Nuclear Reactor Regulation
Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

December 18, 1984
ANPP-31502 WFQ/TFQ

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Post-FDA Proposed CESSAR Changes
File: 84-056-026; G.1.01.10

References: (A) Letter from A. E. Scherer, CE, to D. G. Eisenhut, NRC, dated October 16, 1984 (LD-84-060); Subject: CESSAR Startup Testing.
(B) Letter from A. E. Scherer, CE, to D. G. Eisenhut, NRC, dated December 5, 1984 (LD-84-070); Subject: CESSAR Amendment 10.

Dear Mr. Knighton:

As discussed with the NRC staff, it is APS' understanding that post-Final Design Approval (FDA) proposed CESSAR changes will not be reviewed and approved by the NRC staff, unless a change meets the criteria of the NRC's Standardization Policy. That is, a proposed change to CESSAR will be reviewed and approved if there is an increase in the safety and health of the public.

With this staff position in mind, APS has reviewed all of the proposed CESSAR changes submitted to the staff, including CESSAR Amendment 9, issued on February 27, 1984. With regard to those changes which will not be reviewed on the CESSAR docket due to the Standardization Policy, APS hereby requests that the attached, previously submitted, proposed CESSAR changes be reviewed as proposed PVNGS FSAR changes.

This request is justified because approval of the attached proposed FSAR changes will allow 1) incorporation of improvements into the initial test program, 2) use of the most recent CE secondary water chemistry guidelines, and 3) removal of unwarranted restrictions on the transfer of hydrogens. Your review and approval will also accept 1) the justification for using primary system blowdown and 2) separation analyses with respect to Regulatory Guide 1.75 for Plant Protection System Cabinets.

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Please contact Mr. William Quinn of my staff if you have any questions on this matter.

Very truly yours,

E. E. Van Brunt, Jr.
APS Vice President
Nuclear Production
ANPP Project Director

EEVB/TFQ/mb
Attachments

cc: E. A. Licitra (w/a)
A. C. Gehr (w/a)
R. P. Zimmerman (w/a)

December 18, 1984
ANPP-31502

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, Edwin E. Van Brunt, Jr., represent that I am Vice President, Nuclear Production of Arizona Public Service Company, that the foregoing document has been signed by me on behalf of Arizona Public Service Company with full authority to do so, that I have read such document and know its contents, and that to the best of my knowledge and belief, the statements made therein are true.

Edwin E. Van Brunt, Jr.

Sworn to before me this 18 day of December, 1984.

Notary Public

My Commission Expires:
My Commission Expires April 6, 1987

EXCERPTS FROM CESSAR

AMENDMENT 9

ISSUED FEBRUARY 27, 1984

5.4.13.4 Tests and Inspections

The valves are inspected during fabrication in accordance with ASME III Code requirements.

5.4.13.4.1 Pressurizer Safety Valves

The inlet and outlet portions of the valves are hydrostatically tested with water at the appropriate pressures required by the applicable section of the ASME Code. Set pressure and seat leakage tests are performed with steam using a pro-rated spring. Final set pressure tests are performed with the final springs using either high pressure air or low pressure steam with an hydraulic assist device. Final seat leakage tests are performed prior to shipment with the final springs using either hot air or hot nitrogen. Valve adjustment shall be made to a valve ring setting combination selected to provide stable valve operation on the basis of the EPRI Safety Valve Test Program results.(1) 9

5.4.13.4.2 Main Steam Safety Valves

The inlet portion of the valve is hydrostatically tested with water in accordance with the ASME Code. Set pressure and set leakage tests are performed using steam. Adjustment is made to provide a valve blowdown meeting the requirement specified in Table 5.4.13-2. 9

(1) CEN-227 "Summary Report on the Operability of Pressurizer Safety Relief Valve in C-E Designed Plants", December 1982. 9

TABLE 5.4.13-1

PRESSURIZER SAFETY VALVE PARAMETERS

Property	Parameter	
Design pressure, lb/in.^{2}a	2500	
Design temperature, °F	700	
Fluid	Saturated Steam, 4400 ppm, boron, pH = 4.5 to 10.6	9
Set pressure, lb/in.^{2}a	2500 ± 1%	
Min. capacity, lb/h at accumulation pressure, each	460,000	9
Type	Spring loaded safety-balanced bellows. Enclosed bonnet.	
Orifice area, in.^{2}	4.34	
Accumulation, %	3	
Backpressure		
Max. buildup/max superimposed, lb/in.^{2}g	700/340	9
Minimum blowdown pressure, psia	2040	9
Typical materials		
Body	ASME SA 182, GR. F316	
Disc	ASTM A637, GR. 688	
Nozzle	ASME SA 182, Gr. 347	9

TABLE 5.4.13-2

MAIN STEAM SAFETY VALVE PARAMETERS

Property	Parameter	
Design pressure, $lb/in.^2g$	1375	
Design temperature, °F	575	
Fluid	Saturated Steam	
Set pressure, $lb/in.^2g$	1255, 1290, 1315	9
Min. capacity, lb/h at accumulation pressure	19×10^6 Total (20 Valves)	9
Type	Spring loaded	
Orifice area, $in.^2$	16	
Accumulation, %	3	
Backpressure		
Max. buildup/max superimposed, $lb/in.^2g$	125/0	9
Approx. dry weight, lbs.	1545	
Minimum blowdown pressure, psig	1175	9
Typical materials		
Body	ASME SA 105	
Disc	ASTM A565, GR. 616	
Nozzle	ASME SA 182, GR. F316	

APPENDIX 5A

OVERPRESSURE PROTECTION FOR COMBUSTION ENGINEERING SYSTEM 80 - PRESSURIZED WATER REACTORS

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3.0	CONCLUSIONS	5A-5

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temperature coefficient can vary between zero and -3.5×10^{-4} for various phases of core life. Therefore, a coefficient of zero is chosen to maximize the power/pressure transient.

c. Doppler coefficient of $-.8 \times 10^{-5}$ $\Delta K/K/F$ is used in the loss-of-load analysis. Actual operating coefficients can be expected to range from -1.4×10^{-5} at zero power to $-1. \times 10^{-5}$ $\Delta K/K/F$ at full power. By choosing a relatively small Doppler coefficient, the reduction in reactivity with increasing fuel temperature is minimized, thereby maximizing the rate of power rise.

d. No credit is taken for letdown, charging, pressurizer spray, turbine bypass, or feedwater addition after turbine trip in the loss-of-load analysis. Letdown and pressurizer spray both act to reduce primary pressure. By not taking credit for these systems, the rate of pressurization is increased. By not taking credit for the addition of feedwater, the steam generator secondary inventory will be depleted at a faster rate. This in turn reduces the capability of the steam generator to remove heat from the primary loop, thereby maximizing the rate of primary pressurization. 9

e. The analysis reflects consideration of plant instrumentation error and safety valve setpoint errors. For example, all safety valves are assumed to open at their maximum popping pressure. This extends the period of time before energy can be removed from the system. The reactor trip setpoint errors are always assumed to act in such a manner that they delay reactor trip, again resulting in maximum pressurization.

f. Pressurizer pressure at the onset of the incident is 2200 psi. By using the lower limit of the normal plant operating pressure, the time required to trip the plant on high pressure is increased.

2.2.1 SECONDARY SAFETY VALVE SIZING

The discharge piping serving the secondary safety valves is designed to accommodate rated relief capacity without imposing unacceptable backpressure on the safety valves.

The secondary safety valves are conservatively sized to pass excess steam flow. This limits steam generator pressure to less than 110% of steam generator design pressure during worst case transients. A plant's secondary safety valves consist of three banks of valves with staggered set pressures. The valves are spring-loaded type safety valves procured in accordance with ASME Boiler and Pressure Vessel Code, Section III. 9

Figure 5A-2 depicts the steam generator pressure transient for this worst case loss-of-load incident. As can be seen in Figure 5A-2, the steam generator pressure remains below 110 percent of design pressure during the incident.

2.2.2 PRIMARY SAFETY VALVE SIZING

The reactor drain tank, inlet and discharge piping are sized to preclude unacceptable pressure drops and backpressure which would adversely affect valve operation.

Primary safety valve backpressure is limited by the design pressure of the valve bellows. These bellows prevent any accumulated backpressure from being imposed on the valve spring, thus allowing valve operation at its design setpoint rather than at its setpoint plus backpressure.

The design basis incident for sizing the primary safety valves is a loss of turbine-generator load in which the reactor is not immediately tripped. No credit is taken for any pressure-reducing devices except the primary and secondary safety valves. In reality, the incident would be terminated by a number of reactor trips. These include: 9

a. Steam generator low level trip;

b. High pressurizer pressure trip;

c. Manual trip.

If the high primary pressure trip were to become inoperative, other reactor trips would proceed to shut the reactor down as their setpoints are exceeded.

A series of loss-of-load studies are run with various sizes of primary safety valves. As can be seen in Figure 5A-1, after the safety valve capacity increases to a certain size, additional increase in capacity has negligible effect in reducing the maximum system pressure experiences during the loss-of-load transient. C-E's primary safety valves are chosen so as to minimize the maximum pressure experienced during the loss-of-load transient. The minimum specified safety valve capacity is identified on Figure 5A-1.

Figures 5A-2, 5A-3 and 5A-4 present curves of steam generator pressure, maximum Reactor Coolant System pressure and core power versus time for the worst case loss of turbine-generator load. As can be seen on Figures 5A-2 and 5A-3, the maximum steam generator pressure and reactor coolant loop pressures remain below 110% of design during this worst case transient.

The first, second, and third banks of secondary safety valves open at approximately 3.7, 5, and 6.2 seconds, respectively. The secondary safety valves remove energy from the Reactor Coolant System and thus mitigate the pressure surge. The primary safety valves are conservatively assumed to open at 1 percent above the normal Reactor Coolant System design pressure 5.7 seconds after the initiation of the upset condition.

In the event that a complete loss of load occurs without a simultaneous reactor trip, the protection provided by the high pressurizer pressure trip, primary safety valves and secondary safety valves is sufficient to assure that the integrity of the RCS and main steam system is maintained and that the minimum DNB ratio is not less than 1.19.

2.2.3 ACCEPTABILITY OF SAFETY VALVE BLOWDOWN

2.2.3.1 Background

Full scale, full pressure prototypical testing of pressurixer safety valves was performed by EPRI in 1981.(1) The blowdown settings required to insure stable valve operation during the blowdown from the set pressure were above the 5% setting specified in the ASME Code. In order to insure that the extended blowdown would not adversely affect overpressure protection or plant operation, analyses were performed to evaluate the NSSS response. The analyses described below demonstrate that a blowdown setting, including associated uncertainties, of 18.5% is acceptable.

2.2.3.2 Results of Evaluation

An extended blowdown of the safety valves could result in swelling of the pressurizer liquid level due to flashing and possible liquid carryover through the safety valves. Since the safety valve design specification specifies dry saturated steam flow conditions, it is desirable to show that these conditions are maintained during the extended blowdown. It is also desirable to verify that the RCS remains in a subcooled condition in order that the steam bubble formation in the RCS is precluded.

A computer analysis was performed of the Loss-of-load event with delayed reactor trip, similar to that used in safety valve sizing, except that a conservative 20% safety valve blowdown and initial conditions biased to maximize pressurizer liquid level were assumed. The purpose of this analysis was to determine the pressurizer liquid level response and the RCS subcooling under these conservative conditions. For additional conservatism, an additive adjustment was made to the computer-calculated pressurizer levels on the basis of a very conservative pressurizer model. This model assumed that the initial saturated pressurizer liquid did not mix with the cooler insurge liquid, that the initial liquid remained in equilibrium with the pressurixer steam space, and that the steam which flashed during blowdown remained dispersed in the liquid phase and caused the liquid level to swell. The adjusted pressurizer water level vs time curve showed a maximum of 98%(2) (1730 ft^3), below the safety valve nozzle elevation of 107%, so that dry saturated steam flow to the safety valves is assured throughout the blowdown. The computer analysis also showed that adequate subcooling was maintained in the RCS during the blowdown, so that steam bubble formation is precluded.

(1) CEN-227, "Summary Report on the Operability or Pressurizer Safety Relief Valves in C-E Designed Plants", December 1982.

(2) Water level expressed as the percentage of the distance from the lower Level nozzle to the upper level nozzle.

In addition, the System 80 safety analyses of pressurization events were re-evaluated to determine the impact of assuming an 18.5% blowdown below nominal set pressure (to 2040 psia) for the pressurizer safety valves in lieu of the 5% specified by the ASME Code. The evaluation indicated that, for the FWLB event analysis, which produces the greatest increase in pressurizer level, the increased blowdown would not result in the pressurizer liquid level reaching the safety valve nozzle elevation and thus normal safety valve operation would be assured. Further, subcooling in the RCS was maintained during the blowdown. 9

In summary, analyses show that adequate plant overpressure protection and RCS subcooling are ensured during a blowdown of 18.5% below nominal pressurizer safety valve set pressure.

3.0 CONCLUSIONS

C-E's System 80 pressurized water reactor, steam generators, and Reactor Coolant System are protected from overpressurization in accordance with the guidelines set forth in the ASME Boiler and Pressure Vessel Code, Section III. Peak Reactor Coolant System and Secondary System pressures are limited to 110% of design pressures during worst case loss of turbine-generator load. Overpressure protection is afforded by primary safety valves, secondary safety valves, and the Reactor Protection System.

14.2.5 REVIEW, EVALUATION, AND APPROVAL OF TEST RESULTS

The development of administrative procedures for review, evaluation and approval of test results is the responsibility of the Applicant. Advice and consultation will be provided by Combustion Engineering as appropriate.

Test results shall be recorded as permanent plant records.

14.2.6 TEST RECORDS

An official copy of each completed test procedure, including all required supplemental data, exceptions, conclusions and approval signatures shall be maintained in accordance with the Applicant's administrative controls.

14.2.7 CONFORMANCE OF INITIAL TEST PROGRAMS WITH REGULATORY GUIDES AND INDUSTRY STANDARDS

The intent of the following Regulatory Guides will be followed with the noted differences.

14.2.7.1 Reg. Guide 1.68 Initial Test Programs for Water-Cooled Reactor Power Plants (Revision 0, 11/73).

The following exceptions and/or clarifications address only significant differences between the System 80 test program and the applicable regulatory position. Minor terminology differences, testing not applicable to the plant design, and testing that is part of required surveillance tests will not be addressed. Reference is made to the applicable portion of Regulatory Guide 1.68 (Revision 0, 11/73). 6

14.2.7.1.1 Reference Appendix A, Section B.1.c. 9

This section suggests that rod drop times be measured for all control element assemblies (CEAs) at hot and cold full-flow and no-flow conditions.

The CESSAR CEA drop-time testing is consistent with the recommendations of the regulatory guide; however, tests which do not provide meaningful data will be deleted. As outlined in test summary 14.2.12.3.4, the CEA drop time testing will consist of: 6

a.) One drop of each CEA at cold, maximum permissible flow conditions (2 or 3 reactor coolant pumps) and at hot, full-flow conditions.*

b.) Those CEAs falling outside the two-sigma limit for similar CEAs will be dropped three additional times.

c.) Hot no flow scram insertion rod drops will not be performed for System 80 reactors. C-E has demonstrated that rod drop times under full-flow conditions are more limiting than the drop times under conditions of no-flow. 9

* Only "first-of-a-kind" plants are allowed to be critical at reduced temperature. Thus, if plant operating procedures preclude pulling CEA's prior to achieving HOTZERO POWER conditions, the cold drops will be expected. 6

14.2.12.5 Power Ascension Tests

14.2.12.5.1 Natural Circulation Test*

1.0 OBJECTIVE

1.1 To evaluate natural circulation flow conditions.

2.0 PREREQUISITES

2.1 The reactor is at ≥ 80% power. Previous power history is such that the loop differential temperature ($T_H - T_C$) under natural circulation is not expected to drop below 10°F during the performance of the test. 9

3.0 TEST METHOD

3.1 The reactor coolant pumps are tripped.

3.2 The plant is tripped.

3.3 The natural circulation power to flow ratio is determined.

4.0 DATA REQUIRED

4.1 Conditions of the measurement.

4.1.1 Power.

4.1.2 Previous power history.

4.1.3 RCS temperatures.

4.2 Time dependent information.

4.2.1 RCS temperatures.

4.2.2 Steam generator levels and pressures.

4.2.3 Pressurizer pressure and level.

5.0 ACCEPTANCE CRITERIA

5.1 The natural circulation power to flow ratio is less than 1.0.

14.2.12.5.2 Variable Tavg (Isothermal Temperature Coefficient and Power Coefficient) Test

1.0 OBJECTIVE

1.1 To measure the Isothermal Temperature Coefficient and Power Coefficient of reactivity at selected power levels.

*This test will be performed for the "first-of-a-kind" unit only.

2.2 The RRS, FWCS, SBCS, RPCS, and the pressurizer level and pressure control systems are in automatic operation.

3.0 TEST METHOD

3.1 Load increases and decreases (steps and ramps) in accordance with the C-E Fuel Pre-conditioning Guidelines will be performed at power levels in the 90 to 100% range and with swings in the 50 to 25 to 50% power level. 6

4.0 DATA REQUIRED

4.1 Time dependent data.

4.1.1 Pressurizer level and pressure.

4.1.2 RCS temperatures.

4.1.3 CEA position.

4.1.4 Power level and demand.

4.1.5 Steam generator levels and pressures.

4.1.6 Feedwater and steam flow.

4.1.7 Feedwater temperature. 9

5.0 ACCEPTANCE CRITERIA

5.1 The step and ramp transients demonstrate that the plant performs load changes allowed by C-E's Fuel Pre-conditioning Guidelines and data has been taken that will demonstrate the plant's ability to meet unit load swing design transients. 6 9

14.2.12.5.4 Control Systems Checkout Test

1.0 OBJECTIVE

1.1 To demonstrate that the automatic control systems operate satisfactorily during steady-state and transient conditions.

2.0 PREREQUISITES

2.1 The reactor is operating at the desired conditions.

2.2 The RRS, FWCS, SBCS, RPCS, and the pressurizer level and pressure controls are in automatic operation.

3.0 TEST METHOD

3.1 The performance of the control systems during normal operations, transients and trips will be monitored to demonstrate that the systems are operating satisfactorily.

2.0 PREREQUISITES

2.1 The SBCS, FHCS, RRS, RPCS, and pressurizer pressure and level controls are operable in either manual or automatic modes. | 7

3.0 TEST METHOD

3.1 Performance of the feedwater systems will be monitored during normal operation, transients, and trips. Specifically, the downcomer to economizer transfer will be monitored for noise or vibration due to Water Hammer. | 9

4.0 DATA REQUIRED

4.1.1 Reactor power

4.1.2 RCS temperatures

4.1.3 Pressurizer pressure

4.1.4 Steam generator levels and pressures

4.1.5 Steam and feedwater flows

4.1.6 Feedwater temperature

4.1.7 CEA Position | 9

5.0 ACCEPTANCE CRITERIA

5.1 The main and emergency feedwater systems perform as designated by the system description.

14.2.12.5.18 CPC Verification | 9

1.0 OBJECTIVE

To verify DNBR and Local Power Density (LPD) calculations of the CPCs.

2.0 PREREQUISITES

2.1 The reactor is at the desired power level and CEA configuration with equilbrium Xe.

2.2 The CPCs are operational.

2.3 The incore detector system is operational.

3.0 TEST METHOD

3.1 Specified values are recorded from the CPCs.

3.2 The values for LPD and DNBR obtained from the CPCs are compared with the values calculated for the same conditions using the CPC FORTRAN Simulator.

C-E Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Tel. 203/688-1911
Telex: 99297

POWER SYSTEMS

STN 50-470F

October 16, 1984
LD-84-060

Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: CESSAR Startup Testing

Dear Mr. Eisenhut:

During preparation for startup of the first System 80™ plant, C-E has noted minor modifications which could be made to CESSAR Chapter 14 to facilitate an improved testing procedure. These changes affect only tests performed after fuel loading and do not in any way affect CESSAR's compliance with NRC requirements. These changes, along with a description of each change, are provided in the attachment for NRC review. These changes will be incorporated in a subsequent amendment to CESSAR.

If you have any questions or comments, feel free to call me or Mr. T. J. Collier of my staff at (203) 285-5215.

Very truly yours,

COMBUSTION ENGINEERING, INC.

A. E. Scherer
Director
Nuclear Licensing

AES:las
Attach.
cc: K. Eccleston (USNRC Project Manager)

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14.2.7.1.1: RG 1.68, Appendix A, Section B.1.c (page 14.2-5)

Deleting the cold (260°F) partial flow CEA drops is consistent with experience in previous startups, which showed that the hot, full-flow drops were more limiting. Low temperature criticality is allowed only on first-of-a-kind plants, and then only for short periods of time under close supervision. CEA insertability at cold conditions is still demonstrated during post-core hot functional testing, providing assurance that the CEAs can be tripped, if necessary.

14.2.10.1: Initial Fuel Loading (page 14.2-7a)

The containment evacuation alarm described in the deleted material will not be provided, nor is there any requirement for such a device. Should a situation exist requiring evacuation, the operator could (in the case of Palo Verde is required to) utilize the site public address system and the plant evacuation alarm. As stated in this section, audible count rate indicators will be provided in containment.

14.2.12.3.1: PCHFT Controlling Document (page 14.2-69)

Item 2.1 reflects the possibility that some of the pre-core tests may be postponed to, or rerun during, the post-core hot functional testing. The other changes reflect how the instrumentation is to be calibrated.

14.2.12.3.4: Post-Core CEDM Performance (page 14.2-72)

The test method is updated to reflect the fact that no cold drops are to be performed. Verification of position indication and alarms is not temperature or schedule dependent and can be accomplished at any time, as the change reflects. The change to the required data reflects the fact that the RCS conditions are only a concern for this test during the rod drops.

14.2.12.4.2: CEA Symmetry and Coupling Test (pages 14.2-77a, 14.2-78)

The CEA coupling test is deleted because of a difference in the System 80 design and previous C-E designs. In previous C-E designs, CEAs and extension shafts were uncoupled during each refueling outage (CEAs remained in the core). In the System 80 design, CEAs and extension shafts are not uncoupled (CEAs are withdrawn into the upper guide structure).

14.2.12.4.4: Shutdown and Regulating CEA Group Worth Tests (page 14.2-79)

Testing on the first-of-a-kind plant has been modified to perform the net shutdown measurement at low temperature (approximately 320°F) rather than at HZP conditions. This change is advantageous for the following reasons:

(1) The measurement provides direct verification of the net shutdown worth at relatively cold conditions (shutdown margin following a cooldown). This measurement can be readily done on the first-of-a-kind unit where a low temperature test program is performed.

(2) Less RCS boron dilution is required.

(3) Potential cooldown events during testing while in a highly rodded configuration would have less of a consequence.

The remaining information, e.g., CEA group worths, obtained with the revised measurement approach is essentially equivalent to that obtained with the original test approach. For follow-on plants, the net shutdown measurement is to be performed at 565°F, since low temperature testing is not performed.

14.2.12.4.7: Pseudo Dropped and Ejected CEA Worth Test (page 14.2-83)

The wording is changed to provide flexibility in the testing methodology. The measurements of CEA worths via dilution (CEA insertion), boration (CEA withdrawal) or CEA compensation provide equivalent information. The test method is reworded to clarify the conditions at which testing will be performed.

14.2.12.5.3: Unit Load Transient Test (page 14.2-87)

The test method is updated to reflect the conditions under which the test will be conducted, including the limiting factors.

14.2.12.5.4: Control Systems Checkout Test (pages 14.2-87, 14.2-88)

The test method is reworded to clarify the conditions at whiich testing will be performed. Feedwater temperature is added to the list of monitored parameters (4.1.7). The acceptance criteria are reworded to clarify the criteria to be used for evaluating steady state performance and transient responses.

14.2.12.5.6: Turbine Trip (pages 14.2-89, 14.2-89a)

The Turbine Trip Test and the Unit Load Rejection Test lead to essentially the same plant response. Rather than perform redundant tests, the turbine trip test will be performed with the Reactor Power Cutback System (RPCS) not in-service while the unit load rejection test is performed with the RPCS in-

service. The Data Required section is reworded to specify the parameters to be evaluated against acceptance criteria. Additional key parameters are to be monitored to provide supplemental information but are not evaluated against specific acceptance criteria. The Acceptance Criteria section is reworded to specify the method to be used for evaluating the parameters against which acceptance criteria are applied. Since non-safety parameters monitored during the test are not evaluated against specific acceptance criteria, the second sentence is eliminated.

14.2.12.5.7: Unit Load Rejection Test (pages 14.2-90, 14.2-90a, 14.2-91)

This test will be performed with the RPCS in operation. The summary is reworded in a manner similar to the Turbine Trip Test.

14.2.12.5.11: Xenon Oscillation Control (PLCEA) Test (pages 14.2-93, 14.2-94)

The initial conditions for the test are revised to allow this test to be performed at or above 50% power. The prerequisite that testing at the 80% plateau be completed is not required. The acceptance criteria is reworded to eliminate the phrase "throughout core life", since this requirement cannot be demonstrated directly from the test results. The test data, in conjunction with design analyses, demonstrates that xenon oscillations are readily controllable throughout life.

14.2.12.5.12: "Ejected" CEA Test (page 14.2-95)
14.2.15.5.13: "Dropped" CEA Test (page 14.2-96)

The rewording of the acceptance criteria clarifies the procedure to be used for evaluating the test results. The rewording does not change the intended acceptance criteria.

14.2.12.5.14: Steady State Core Performance Test (pages 14.2-96, 14.2-97)

The objective is reworded to coincide with the primary reason for performing the test. Objective 1.1 is deleted since specific acceptance criteria are not applied for this purpose. The Test Method and Data Required sections are reworded to more clearly specify the way the test will be performed. Acceptance Criteria 5.1 is not required as the COLSS and CPC systems adequately monitor DNBR and LPD limits during power escalation. Acceptance Criteria 5.2 is reworded to specify that core peaking factors are also evaluated against specific acceptance criteria.

14.2.12.5.15: Intercomparison of PPS, CPC and PMS Inputs (page 14.2-99)

The rewording reflects the proper terminology for this parameter, i.e., the temperature shadowing factor. This measurement is planned for both first-of-a-kind and follow-on units, so the asterisked footnote is deleted.

14.2.12.5.17: Main and Emergency Feedwater Systems Test (page 14.2-99a)

This page was missing from Amendment 9, February 27, 1984.

14.2.12.5.18: CPC Verification (page 14.2-101)

Incore detector maps (Section 4.5) are not required for this test so this requirement is deleted. The Acceptance Criteria section is reworded to clarify the procedure for applying the criteria.

14.2.12.5.19: Steam Bypass Valve Capacity Test (pages 14.2-101, 14.2-102)

The test description is altered to reflect capacity testing of each ADV and SBCS valve individually. Individual valve capacities are required to show that the valve capacities assumed in Chapter 15 (Safety Analysis) are conservative. Prerequisites are changed to delete the requirement for automatic SBCS operation, because individual valve modulation (open and close) is not possible in automatic control.

Table 14.2-1: Low Power Physics Tests

The table is modified to be consistent with the revised test summaries (described above).

Table 14.2-2: Power Ascension Test

The table is modified to be consistent with the test summaries and to reflect the planned testing approach. The footnote on the coefficient measurements is added to clarify that the test is performed with CEA movement and must be performed at a power level which allows the required CEA motion based on margin considerations.

Table 14.2-7: Physics (Steady State) Test Acceptance C iteria

The acceptance criteria for net shutdown worth, and dropped and ejected CEA worths, were inadvertently omitted. These are added in this amendment. Dropped and ejected CEA worths, and power distribution comparisons, are not required on follow-on units. This change makes Table 14.2-7 consistent with Tables 14.2-1 and 14.2-2. Other addtions are added for clarification and do not affect the established acceptance criteria.

MODIFICATIONS

TO CHAPTER 14

OF CESSAR-F

14.2.5 REVIEW, EVALUATION, AND APPROVAL OF TEST RESULTS

The development of administrative procedures for review, evaluation and approval of test results is the responsibility of the Applicant. Advice and consultation will be provided by Combustion Engineering as appropriate.

Test results shall be recorded as permanent plant records.

14.2.6 TEST RECORDS

An official copy of each completed test procedure, including all required supplemental data, exceptions, conclusions and approval signatures shall be maintained in accordance with the Applicant's administrative controls.

14.2.7 CONFORMANCE OF INITIAL TEST PROGRAMS WITH REGULATORY GUIDES AND INDUSTRY STANDARDS

The intent of the following Regulatory Guides will be followed with the noted differences.

14.2.7.1 Reg. Guide 1.68 Initial Test Programs for Water-Cooled Reactor Power Plants (Revision 0, 11/73).

The following exceptions and/or clarifications address only significant differences between the System 80 test program and the applicable regulatory position. Minor terminology differences, testing not applicable to the plant design, and testing that is part of required surveillance tests will not be addressed. Reference is made to the applicable portion of Regulatory Guide 1.68 (Revision 0, 11/73).

14.2.7.1.1 Reference Appendix A, Section B.1.c.

This section suggests that rod drop times be measured for all control element assemblies (CEAs) at hot and cold full-flow and no-flow conditions.

The CESSAR CEA drop-time testing is consistent with the recommendations of the regulatory guide; however, tests which do not provide meaningful data will be deleted. As outlined in test summary 14.2.12.3.4, the CEA drop time testing will consist of:

a.) One drop of each CEA ~~at cold, maximum permissible flow conditions (0 or 3 reactor coolant pumps) and~~ at hot, full-flow conditions.

b.) Those CEAs falling outside the two-sigma limit for similar CEAs will be dropped three additional times.

c.) Hot no flow scram insertion rod drops will not be performed for System 80 reactors. C-E has demonstrated that rod drop times under full-flow conditions are more limiting than the drop times under conditions of no-flow.

d.) The CEA drop time test at 260°F Plateau was eliminated since Only "first-of-a-kind" plants are allowed to be critical at reduced temperature. Thus, if plant operating procedures preclude pulling CEA's prior to achieving HOT ZERO POWER conditions, the cold drops will be expected.
the hot, full-flow conditions are more bounding and since criticality is not allowed below 500°F except for a short period of time during low power physics testing.

one temporary channel and one permanent channel will be equipped with audible count rate indicators in two locations, temporary in the containment and permanent in the main control room. ~~A containment evacuation alarm is coupled to the permanent wide-range nuclear channels with a setpoint at five times the latest equilibrium count rate to provide automatic indication of high count rate during the fuel loading operation.~~

Continuous area radiation monitoring will be provided during fuel handling and fuel loading operations. Permanently installed radiation monitors display radiation levels in the main control room and will be monitored by licensed operators.

Fuel assemblies, together with inserted components, will be placed in the reactor vessel one at a time according to a previously established and approved sequence which was developed to provide reliable core monitoring with minimum possibility of core mechanical damage. The initial fuel loading procedure will include detailed instructions which will prescribe successive movements of each fuel assembly from its initial position in the storage racks to its final position in the core. The procedures will establish a system and a requirement for verification of each fuel assembly movement prior to proceeding with the next assembly. Multiple checks will be made for fuel assembly and inserted component serial numbers at successive transfer points to guard against possible inadvertent exchanges or substitution.

At least two fuel assemblies containing neutron sources will be placed into the core at appropriate specified points in the initial fuel loading procedure to ensure a neutron population large enough for adequate monitoring of the core. As each fuel assembly is loaded, at least two separate inverse count rate plots will be maintained to ensure that the extrapolated inverse count rate ratio behaves as would be expected. In addition, nuclear instrumentation will be monitored to ensure that the "just loaded" fuel assembly does not excessively increase the count rate. The results of each loading step will be reviewed and evaluated before the next prescribed step is started.

14.2.10.1.1 Safe Loading Criteria

Criteria for the safe loading of fuel require that loading operations stop immediately if:

a.) The neutron count rate from either temporary nuclear channel unexpectedly doubles during any single loading step, excluding anticipated change due to detector and/or source movement or spatial effects (i.e., fuel assembly coupling source with a detector), or

b.) The neutron count rate on any individual nuclear channel increases by a factor of five during any single loading step, excluding anticipated changes due to detector and/or source movement or spatial effects (i.e., fuel assembly coupling source with a detector).

14.2.12.3 Postcore Hot Functional Tests

14.2.12.3.1 Postcore Hot Functional Test Controlling Document

1.0 OBJECTIVE

To demonstrate the proper integrated operation of plant primary, secondary, and auxiliary systems with fuel loaded in the core.

2.0 PREREQUISITES

2.1 All precore hot functional testing has been completed, as required.

2.2 Fuel loading has been completed.

2.3 All permanently installed instrumentation on systems to be tested is available and calibrated, in accordance with technical specifications and test procedures.

2.4 All necessary test instrumentation is available and calibrated, in accordance with technical specifications and test procedures.

2.5 All cabling between the CEDM's and the CEDM control system is connected.

2.6 Steam generators are in wet layup in accordance with the NSSS chemistry manual.

2.7 RCS has been borated to the proper concentration.

3.0 TEST METHOD

3.1 Specify plant conditions and coordinate the execution of the related postcore hot functional test appendices.

4.0 DATA REQUIRED

4.1 As specified by the individual postcore hot functional test appendices.

5.0 ACCEPTANCE CRITERIA

5.1 Integrated operation of the primary, secondary, and related auxiliary systems is in accordance with the CESSAR descriptions.

5.2 As specified by the individual postcore hot functional test appendices.

14.2.12.3.4 Postcore Control Element Drive Mechanism Performance

1.0 OBJECTIVE

1.1 To demonstrate the proper operation of the CEDM's and CEA's under HOT SHUTDOWN and Hot, Zero Power conditions.

1.2 To verify proper operation of the CEA position indicating system and alarms.

1.3 To measure CEA drop times.

2.0 PREREQUISITES

2.1 The CEDMCS precore performance test has been completed.

2.2 All test instrumentation is available and calibrated.

2.3 Plant Monitoring System is operational.

2.4 The CEDM cooling system is operational.

2.5 CEDM coil resistances have been measured.

3.0 TEST METHOD

3.1 Perform the following at HOT SHUTDOWN conditions:

3.1.1 Withdraw and insert each CEA ~~while recording position indications and alarms.~~ to verify proper operation of CEDM.

~~3.1.2 Measure and record drop time for each CEA.~~

~~3.1.3 Perform three additional measurements of drop time for each of those CEA's falling outside the two-sigma limit for similar CEA's.~~

3.2 Perform the following at hot, zero power conditions:

3.2.1 Withdraw and insert each CEA ~~while recording position indications and alarms.~~ to verify proper operation of CEDM.

3.2.2 Measure and record drop time for each CEA.

3.2.3 Perform three measurements of drop time for each of those CEA's falling outside the two-sigma limit for similar CEA's.

3.3 Perform the following at any time:

3.3.1 Withdraw and insert each CEA while recording position indications and alarms.

4.0 DATA REQUIRED

4.1 CEA drop time.

4.2 RCS temperature and pressure to be taken during measurement and recording of drop time for each CEA.

4.3 CEA position and alarm indications.

14.2.12.4.2 CEA Symmetry ~~and Coupling~~ Test~~**~~

1.0 OBJECTIVE

1.1 To demonstrate that no loading or fabrication errors that result in measurable CEA worth asymmetries have occurred.

~~1.2 To demonstrate proper coupling of each CEA to its drive mechanism.~~

~~**For the "first of a kind" unit the coupling checks will be performed at HOT SHUTDOWN conditions and the symmetry checks will be performed at 565°F. For "follow-on" units only the symmetry check at 565°F is performed.~~

2.0 PREREQUISITES

2.1 The reactivity computer is in operation.

2.2 The reactor is critical at the desired conditions with the controlling CEA group partially inserted and in manual control.

3.0 TEST METHOD

~~3.1 CEA Coupling Check (HOT SHUTDOWN - "first-of-a-kind")~~

~~3.1.1 A selected CEA is inserted sufficiently to produce a small negative reactivity insertion and then is withdrawn.~~

~~3.1.2 Step 3.1.1 is repeated for the remainder of the CEAs.~~

3.2 CEA Symmetry Test (hot, zero power conditions - 565°F, 2250 psia)

3.2.1 The first CEA of a symmetric group is fully inserted with all remaining CEAs withdrawn except the controlling group, which is positioned for zero reactivity.

3.2.2 The inserted CEA is withdrawn while another CEA in the symmetric group is inserted and the differences in worth (net reactivity) of the CEAs is determined from the reactivity computer.

3.2.3 The remainder of the CEAs in the symmetric group are sequentially swapped until the relative worths of each CEA in the symmetric group has been determined.

3.2.4 Repeat steps 3.2.1 - 3.2.3 for the remainder of the groups.

4.0 DATA REQUIRED

4.1 Conditions of ~~the~~ measurement.

4.1.1 RCS temperature.

4.1.2 Pressurizer pressure.

4.1.3 Boron concentration.

4.2 Time dependent data.

4.2.1 CEA position.

4.2.2 Reactivity computer traces.

5.0 ACCEPTANCE CRITERIA

5.1 The relative worth of symmetric CEAs are within the acceptance criteria specified in Table 14.2-7.

~~5.2 All CEAs are demonstrated to be coupled.~~

14.2.12.4.4 Shutdown and Regulating CEA Group Worth Test

1.0 OBJECTIVE

1.1 To determine regulating ~~(including PIPEAT)~~ and shutdown CEA group worths necessary to demonstrate shutdown margin (i.e., worth of all CEA's less the highest worth CEA).

1.2 To demonstrate that the shutdown margin is adequate.

2.0 PREREQUISITES

2.1 The reactor is critical.

2.2 The reactivity computer is operating.

3.0 TEST METHOD

3.1 ~~HOT SHUTDOWN CONDITIONS~~ A 320°F AND SHUTDOWN measurement of regulating CEA groups down to the ~~ZERO POWER dependent insertion limit (HZPDIL)~~ NET SHUTDOWN CONFIGURATION (for "first-of-a-kind" plant only).*

3.1.1 The CEA group worths will be measured by dilution/boration of the RCS.

3.2 Hot, zero power measurement of regulating ~~and shutdown~~ CEA groups.*

3.2.1 The CEA group worths will be measured by dilution/boration of the RCS.

3.2.2 Where dilution/boration is not feasible, worths may be determined by CEA drop and/or by use of alternate CEA configurations.

4.0 DATA REQUIRED

4.1 Conditions of the measurement.

4.1.1 RCS temperature.

4.1.2 Pressurizer pressure.

4.1.3 CEA configuration.

4.1.4 Boron concentration.

4.2 Time dependant information.

4.2.1 Reactivity variation (strip chart).

4.2.2 CEA positions.

* ON "FOLLOW-ON" UNITS THE NET SHUTDOWN MEASUREMENT IS MADE AT 565°F

14.2.12.4.7 Pseudo Dropped and Ejected CEA Worth Test*

1.0 OBJECTIVE

1.1 To measure the worth of the "dropped" CEA.

1.2 To measure the worth of the "ejected" CEA from the zero power dependent insertion limit (ZPDIL).

2.0 PREREQUISITES

2.1 Reactor critical at hot, zero power conditions with appropriate CEA configurations.

2.2 The reactivity computer is in operation.

3.0 TEST METHOD

3.1 Pseudo worst "dropped" CEA measurement

3.1.1 The pseudo worst and next worst "dropped" CEA worths are established on the basis of predictions and verified during the symmetry check.

3.1.2 The worths of the worst and next worst dropped CEAs are then measured by ~~boron~~ dilution/boration and/or CEA compensation.

3.2 Pseudo worst "dropped" PLCEA and worst "dropped" PLCEA subgroup measurement.

3.2.1 The pseudo worst "dropped" PLCEA and worst "dropped" PLCEA subgroups are established by prediction.

3.2.2 The worths of the worst single PLCEA and PLCEA subgroup are measured by boron dilution/boration and/or CEA compensation.

3.3 Pseudo worst "ejected" CEA measurement

3.3.1 The worth of the pseudo worst "ejected" CEA is established by means of a prediction.

3.3.2 The worths of the worst and next worst "ejected" CEAs are measured by boration ~~for the ZPDIL CEA configuration~~. dilution and/or CEA compensation from the ZPIL CEA configuration.

4.0 DATA REQUIRED

4.1 Conditions of the measurement

4.1.1 RCS temperature

*This test will be performed only on the "first-of-a-kind" plant.

2.2 The RRS, FWCS, SBCS, RPCS, and the pressurizer level and pressure control systems are in automatic operation.

3.0 TEST METHOD

3.1 Load increases and decreases (steps and ramps) in accordance with the C-E Fuel Pre-conditioning Guidelines and as allowed by the RRS will be performed at power levels in the ~~90~~ 75 to ~~100%~~ 95 range ~~and with swings in the 50 to 25 to 50% power level~~ set in the 25 to 50% power range.

4.0 DATA REQUIRED

4.1 Time dependent data.

4.1.1 Pressurizer level and pressure.

4.1.2 RCS temperatures.

4.1.3 CEA position.

4.1.4 Power level and demand.

4.1.5 Steam generator levels and pressures.

4.1.6 Feedwater and steam flow.

4.1.7 Feedwater temperature.

5.0 ACCEPTANCE CRITERIA

5.1 The step and ramp transients demonstrate that the plant performs load changes allowed by C-E's Fuel Pre-conditioning Guidelines and data has been taken that will demonstrate the plant's ability to meet unit load swing design transients.

14.2.12.5.4 Control Systems Checkout Test

1.0 OBJECTIVE

1.1 To demonstrate that the automatic control systems operate satisfactorily during steady-state and transient conditions.

2.0 PREREQUISITES

2.1 The reactor is operating at the desired conditions.

2.2 The RRS, FWCS, SBCS, RPCS, and the pressurizer level and pressure controls are in automatic operation.

3.0 TEST METHOD

3.1 The performance of the control systems during ~~normal operations, power~~ steady state and transient conditions ~~transients and trips~~ will be monitored to demonstrate that the systems are operating satisfactorily.

4.0 DATA REQUIRED

4.1 Time dependent data.

4.1.1 Pressurizer level and pressure.

4.1.2 RCS temperatures.

4.1.3 CEA position.

4.1.4 Power level and demand.

4.1.5 Steam generator levels and pressures.

4.1.6 Feedwater and steam flow.

4.1.7 Feedwater temperature.

5.0 ACCEPTANCE CRITERIA

5.1 The control systems maintain the reactor power, RCS temperature, pressurizer pressure and level, and steam generator levels and pressures within their control bands during ~~both~~ steady state ~~and transient operation~~ operation and are capable of returning these parameters to within their control bands in response to transient operation.

14.2.12.5.5 Reactor Coolant and Secondary Chemistry and Radiochemistry Test

1.0 OBJECTIVE

1.1 To conduct chemistry tests at various power levels with the intent of gathering corrosion data and determining activity buildup.

1.2 To verify proper operation of the process radiation monitor.

1.3 To verify the adequacy of sampling and analysis procedures.

2.0 PREREQUISITES

2.1 The reactor is stable at the desired power level.

2.2 Sampling systems for the RCS and CVCS are operable.

3.0 TEST METHOD

3.1 Samples will be collected from the RCS and secondary system at various power levels and analyzed in the laboratory using applicable sampling and analysis procedures.

3.2 Samples will be collected at the process radiation monitor at various power levels, analyzed in the laboratory, and compared with the process radiation monitor to verify proper operation.

4.0 DATA REQUIRED

4.1 Conditions of the measurement.

4.1.1 Power.

4.1.2 RCS temperature.

4.1.3 Boron concentration.

4.1.4 Core average burnup.

4.2 Samples for measurement of gross activities and/or isotopic activities.

5.0 ACCEPTANCE CRITERIA

5.1 Measured activity levels are within their limits.

5.2 The process radiation monitors agree with the laboratory analyses within measurement uncertainties.

5.3 Procedures for sample collection and analysis are verified.

14.2.12.5.6 Turbine Trip Test

1.0 OBJECTIVE

1.1 To demonstrate that the plant responds and is controlled as designed following a 100% turbine trip without RPCS in service.

2.0 PREREQUISITES

2.1 The reactor is operating above 95% power.

2.2 The SBCS, FWCS, RRS, ~~RPCS~~ and pressurizer pressure and level control systems are in automatic operation.

2.3 The RPCS is in Auto Actuate Out of Service.

3.0 TEST METHOD

3.1 The turbine is tripped.

3.2 The plant behavior is monitored to assure that the RRS, SBCS, FWCS, ~~RPCS~~, and pressurizer pressure and level control systems maintain the NSSS within operating limits.

4.0 DATA REQUIRED

4.1 Power level prior to trip.

4.2 The following acceptance criteria parameters are monitored prior to and throughout the transient.

4.2.1 Pressurizer pressure, and level, ~~and spray flow~~

4.2.2 RCS Hot Leg temperature ~~and pressure~~.

4.2.3 SG pressure ~~and level.~~

~~4.2.4 SBC steam flow and header pressure.~~

~~4.2.5 Feedwater systems (main and auxiliary) feed flow and enthalpy~~.

~~4.2.6 Core power~~.

~~4.2.7 Charging and letdown flows~~.

~~4.2.8 RPS performance and CEA position.~~

4.3 Additional key plant parameters will be monitored for base line data.

5.0 ACCEPTANCE CRITERIA

5.1 The test will be evaluated against single valued acceptance limits for those safety parameters which approach a safety limit. ~~In addition, the time dependent RCS temperatures and pressure as well as SG levels and pressures will be compared to expected values~~.

14.2.12.5.7 Unit Load Rejection Test

1.0 OBJECTIVE

1.1 To demonstrate that the ~~NSSS can sustain a 100% load rejection.~~ plant responds and is controlled as designed following a 100% load rejection with RPCS in service.

2.0 PREREQUISITES

2.1 The reactor is operating above 95% power.

2.2 The SBCS, FWCS, RRS, CEDMCS, RPCS, and pressurizer pressure and level control are in automatic operation.

3.0 TEST METHOD

3.1 A breaker(s) is tripped so as to subject the turbine to the maximum credible overspeed condition.

3.2 The plant behavior is monitored to assure that the RRS, CEDMCS, SBCS, RPCS, FWCS, and pressurizer pressure and level control systems maintain the monitored parameters.

4.0 DATA REQUIRED

4.1 Plant condition prior to trip.

4.2 The following acceptance criteria parameters are monitored prior to and throughout the transient.

4.2.1 Pressurizer pressure, and level ~~and spray flow~~.

4.2.2 RCS Hot Leg temperatures ~~and pressure~~.

4.2.3 SG pressure, ~~and level~~.

~~4.2.4 SBC steam flow and header pressure.~~

4.3 Additional key plant parameters will be monitored for baseline data.

~~4.2.5 Feedwater systems (main and auxiliary) feed flow and enthalpy.~~

~~4.2.6 Core power.~~

~~4.2.7 Charging and letdown flows.~~

~~4.2.8 RPS performance and CEA position.~~

5.0 ACCEPTANCE CRITERIA

5.1 The test will be evaluated against single valued acceptance limits for those safety parameters which approach a safety limit. ~~In addition, the time dependent RCS temperature and pressure as well as SG levels and pressures will be compared to expected values.~~

14.2.12.5.8 Shutdown from Outside the Control Room Test

1.0 OBJECTIVE

1.1 To demonstrate that the plant can be maintained in HOT STANDBY from outside the control room following a reactor trip.

2.0 PREREQUISITES

2.1 The reactor is operating at $\geq$ 10% of rated power.

2.2 The capability to cooldown on the shutdown cooling systems has been demonstrated during pre and post core hot functional tests.

2.3 The remote shutdown panel instrumentation is operating properly.

2.4 The communication systems between the control room and remote shutdown location has been demonstrated to be operational.

2.5 The remote shutdown instrumentation controls and systems have been preoperationally tested.

3.0 TEST METHOD

3.1 The operating crew evacuates the control room (standby crew remains in the control room).

3.2 The reactor is tripped from outside the control room.

3.3 The reactor is brought to HOT STANDBY by the operating crew from outside the control room and is maintained in this condition for at least 30 minutes.

4.0 DATA REQUIRED

4.1 Time dependent data.

2.2 Results of the radiation surveys performed at zero power conditions are available.

3.0 TEST METHOD

3.1 Measure gamma and neutron dose rates at 20, 50, 80 and 100% power levels.

4.0 DATA REQUIRED

4.1 Power level.

4.2 Gamma dose rates in the accessible locations.

4.3 Neutron dose rates in the accessible locations.

5.0 ACCEPTANCE CRITERIA

5.1 Accessible areas and occupancy times during power operation have been defined.

14.2.12.5.11 Xenon Oscillation Control (PLCEA) Test*

1.0 OBJECTIVE

1.1 To demonstrate a technique for damping xenon oscillations.

2.0 PREREQUISITES

~~2.1~~ ~~Testing at 80% power has been completed.~~

2.2[1] The reactor is ~~at approximately 65% power with equilibrium xenon and the PLCEAs inserted.~~ greater than or equal to 50% power.

2.2[2] The COLSS and the incore detector system are in operation.

3.0 TEST METHOD

3.1 A free oscillation is establised.

3.2 The PLCEA's/or CEA's are used to dampen the oscillation.

4.0 DATA REQUIRED

4.1 Reactor conditions.

4.1.1 Power level.

4.1.2 Boron concentration.

*This test will be performed only on the "first-of-a-kind" plant.

4.1.3 RCS temperatures.

4.1.4 Burnup.

4.1.5 CEA position.

4.2 Time dependent data.

4.2.1 Incore detector maps.

4.2.2 Excore detector information.

4.2.3 PLCEA's and CEA position.

5.0 ACCEPTANCE CRITERIA

5.1 The technique necessary to damp xenon oscillations ~~throughout core life~~ using the PLCEAs and/or CEA's has been demonstrated.

14.2.12.5.12 "Ejected" CEA Test*

1.0 OBJECTIVE

1.1 To determine the power distribution associated with the pseudo CEA ejection from the full power dependent insertion limit (FPDIL) CEA configuration.

2.0 PREREQUISITES

2.1 Testing at 80% power has been completed.

2.2 The reactor is at approximately 50% power with equilibrium conditions and with the CEAs at the FPDIL.

2.3 The incore detector system is in operation.

3.0 TEST METHOD

3.1 The "worst" case CEA (selected by calculation) is fully withdrawn.

3.2 Incore detector maps are taken before and after withdrawal of the static "ejected" CEA.

3.3 The next worst "ejected" CEA is withdrawn while inserting the previous CEA.

3.4 An incore detector map is taken.

3.5 The CEAs are returned to normal configuration.

*This test will be performed only on the "first-of-a-kind" plant.

4.0 DATA REQUIRED

4.1 Conditions of the measurement.

4.1.1 Boron concentration.

4.1.2 Burnup.

4.2 Time dependent data.

4.2.1 Power.

4.2.2 Incore and excore detector readings.

4.2.3 RCS temperature.

4.2.4 CEA position.

5.0 ACCEPTANCE CRITERIA

5.1 ~~The differences between the measured and predicted ratios of the post-ejected CEA to pre-ejected CEA power distributions are the normalized incore detector signals are~~ within the acceptance band specified in Table 14.2-7. ~~of those for similarly normalized predicted values.~~

14.2.12.5.13 Dropped CEA Test*

1.0 OBJECTIVE

1.1 To determine the power distribution resulting from a "dropped" CEA.

2.0 PREREQUISITES

2.1 Testing at 80% power has been completed.

2.2 The reactor is at approximately 50% power with equilibrium conditions for the desired CEA configuration.

2.3 The incore detector system is in operation.

3.0 TEST METHOD

3.1 A full length CEA is selected, based on calculations, which will best verify the dropped rod assumptions used in the safety analyses.

3.1.1 The selected CEA is rapidly inserted to the full-in position.

3.1.2 The CEA remains inserted for a preselected time.

3.1.3 Excore and incore instrument signals are recorded before and after the CEA insertion.

*This test will be performed only on the "first-of-a-kind" plant.

3.2 PLCEA

3.2.1 The PLCEA, selected as prescribed in 3.1.1, is rapidly inserted to the full in position.

3.2.2 The PLCEA remains inserted for a preselected time.

3.2.3 Excore and incore instrument signals are recored before and after the CEA insertion.

4.0 DATA REQUIRED

4.1 Conditions of the measurement.

4.1.1 Boron concentration.

4.1.2 Burnup.

4.2 Time dependent data.

4.2.1 Power.

4.2.2 Incore and excore detector readings.

4.2.3 RCS temperatures.

4.2.4 CEA position.

5.0 ACCEPTANCE CRITERIA

5.1 ~~The difference between~~ the measured and predicted ratios of the post-dropped CEA to pre-~~dropped~~ CEA power distributions are ~~The measured normalized power distributions agree with predictions~~ within the acceptance ~~criteria~~ band specified in Table 14.2-7.

14.2.12.5.14 Steady State Core Performance Test

1.0 OBJECTIVE

~~1.1 To monitor NSSS and overall plant performance and establish a data base for future use.~~

1.~~1~~ To determine core power distributions using incore instrumentation.

2.0 PREREQUISITES

2.1 The reactor is operating at the desired power level and CEA configuration with equilibrium Xe.

2.2 The incore instrumentation system is in operation.

3.0 TEST METHOD

3.1 Selected plant computer outputs and CPC outputs are recorded. ~~and console instrument readings are recorded.~~

3.2 Reactor power is determined by performing a heat balance.

3.3 The core power distribution is obtained using the incore detectors.

4.0 DATA REQUIRED

4.1 Conditions of the test.

4.1.1 Reactor power.

4.1.2 CEA positions.

4.1.3 Boron concentration.

4.1.4 Core average burnup.

4.1.5 Selected plant computer outputs and CPC outputs.

~~4.1.6 Selected console instrument readings;~~

4.1.~~X~~6 Incore detector maps.

5.0 ACCEPTANCE CRITERIA

~~5.1 A data base has been established at the plateau and reviewed to determine that DNBR and LPD limits will not be exceeded during escalation to the next power plateau.~~

5.~~X~~1 Agreement between the predicted and measured power distributions within the acceptance criteria specified in Table 14.2-7. and core peaking factors are

14.2.12.5.15 Intercomparison of PPS, Core Protection Calculator (CPC), and PMS Inputs

1.0 OBJECTIVE

1.1 To verify that process variable inputs/outputs of the PPS, the CPCs, the PMS, and the console instruments are consistent.

2.0 PREREQUISITES

2.1 The plant is operating at the desired conditions.

2.2 All CPCs and CEACs, and the PMS are operable.

3.0 TEST METHOD

3.1 Planar radial peaking factors are verified for various CEA configurations by comparison of the CPC values with values measured with the incore detector system.

3.2 The CEA shadowing factors are verified by comparing excore detector responses for various CEA configurations with the unrodded excore responses.

3.3 The shape annealing factors are measured by comparing incore power distributions and excore detector responses during a free Xe oscillation.

3.4 The temperature ~~annealing~~ shadowing factors are verified by comparing core power and excore detector responses for various RCS temperatures.

4.0 DATA REQUIRED

4.1 Conditions of the measurement.

4.1.1 Power.

4.1.2 Burnup.

4.2 Time dependent data.

4.2.1 Incore and excore detector readings.

4.2.2 CEA position.

4.2.3 RCS temperatures.

5.0 ACCEPTANCE CRITERIA

5.1 Measured radial peaking factors determined from incore flux maps are no higher than the corresponding values used in the CPCs.

5.2 The CEA shadowing factors, and temperature ~~annealing~~ shadowing factors used in the CPCs agree within the acceptance criteria specified in the CPC test requirements.*

5.3 The shape annealing matrix have been measured and the boundary point power correlation constants used in the CPCs are within the limits specified by the test requirements.**

~~*This test will be performed only on the "first-of-a-kind" plant.~~

**As specified in the appropriate revisions or supplements of ~~CEN-63A~~ CEN-235(V).

14.2.12.5.17 Main and Emergency Feedwater Systems Test

1.0 OBJECTIVE

1.1 To demonstrate that the operation of the main feedwater and emergency feedwater systems during Hot Standby, Startup and other normal operations, transients, and plant trips is satisfactory. 7

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(missing page from Amendment 9, February 27, 1984)

4.0 DATA REQUIRED

4.1 Reactor power.

4.2 CEA positions.

4.3 Boron concentration.

4.4 Specified CPC inputs, outputs, and constants.

~~4.5 Incore detector maps.~~

5.0 ACCEPTANCE CRITERIA

5.1 The values of DNBR and LPD ~~obtained from~~ calculated by the CPCs are ~~within the acceptance criteria as delineated in CEN 63 A and its updates.~~ consistant with the values calculated by the CPC Fortran code.

14.2.12.5.19 Steam Bypass Valve Capacity Test

1.0 OBJECTIVE

1.1 To demonstrate that the maximum steam flow capacity of ~~a single~~ each atmospheric steam dump valve upstream of the main steam isolation valves is less than that assumed for the safety analysis.

1.2 To measure the capacity of ~~the low capacity~~ each steam bypass valve individually to determine that the ~~high~~ capacity ~~turbine by-pass valves~~ of each steam bypass valve is less than the value used in the ~~safety~~ analysis.

2.0 PREREQUISITES

2.1 The reactor power is $\geq$ 15% full power.

~~2.2 The SBCS is in automatic operation and is bypassing steam to the condenser.~~

2.~~X~~2 Control systems are in automatic where applicable.

2.~~X~~3 The operation of the atmospheric steam ~~drum~~ dump, turbine by-pass and shutdown cooling system have been demonstrated as part of the HOT FUNCTIONAL testing.

3.0 TEST METHOD

3.1 The individual steam flows through each of the atmospheric dump valves upsteam of the MSIVs are measured.

3.2 The capacit~~ies~~y of ~~selected~~ each steam bypass valve~~s~~ ~~are~~ is measured.

4.0 DATA REQUIRED

4.1 Reactor power.

4.2 RCS temperatures.

4.3 Pressurizer pressure.

4.4 Steam generator levels and pressure.

4.5 Steam dump and bypass valve positions.

5.0 ACCEPTANCE CRITERIA

5.1 The capacities of the individual steam dump valves are less than the values used in the safety analysis but greater than the values required for a safe cooldown.

5.2 The capacities of ~~selected~~ each steam bypass valves ~~have~~ as been measured, and the capacity of each steam bypass valve is less than the value used in the safety analysis.

14.2.12.5.20 Incore Detector Test

1.0 OBJECTIVE

1.1 To verify conversion of the fixed incore detector signals to voltages for input to the plant computer.

1.2 To collect baseline performance data for the movable incore detector system.

2.0 PREREQUISITES

2.1 The reactor is at the specified power level and conditions.

2.2 The plant computer is operable.

2.3 The incore detector system is operable.

3.0 TEST METHOD

3.1 Fixed incore detector signal verification.

3.1.1 Amplifier output signals are measured based on test input signals.

3.1.2 Group symmetric instrument signals are measured.

3.2 Data is recorded from the movable incore detectors during core traverses.

4.0 DATA REQUIRED

4.1 Reactor power.

4.2 CEA position.

TABLE 14.2-1

LOW POWER PHYSICS TESTS

Test Title	First-of-a-kind*	Follow-On Units**
Low Power Biological Shield Survey Test	320°F/565°F	565°F
~~CEA Coupling~~ CEA Symmetry Test	~~320°F~~ 565°F	565°F
Isothermal Temperature Coefficient Test	320°F-565°F	565°F
Regulating CEA Group Worth Test	320°F & 565°F	565°F
Shutdown CEA Group Worth Test	~~565°F~~ 320°F	565°F
Differential Boron Worth Test	320°F & 565°F	565°F
Critical Boron Concentration Test	320°F-565°	565°F
Pseudo Dropped and Ejected CEA Worth Test	565°F	N/A

* An expanded test program is conducted for the "first-of-a-kind" in order to validate the design, the design methods, and the safety analysis assumptions.

~~On the "first-of-a-kind" plant the CEA coupling check is performed at 320°F, and the CEA symmetry test is performed at 565°F.~~

** Reduced testing is contingent upon the demonstration that "Follow-On" plants behave in an identical manner as the First-Of-A-Kind plant through conformance with the Acceptance Criteria given in Table 14.2-7.

TABLE 14.2-2

(Sheet 1 of 2)

POWER ASCENSION TEST

Test Title	First-of-a-Kind*	Follow-On Units**
Natural Circulation Test	*** ≥ 80%	N/A
Variable Tavg (Isothermal Temperature Coefficient & Power Coefficient) Test	20, 50, 80, 100%	50 & 100%
Unit Load Transient Test	50, 100% [redacted]	50, 100% [redacted]
Control Systems Checkout Test	20, 50, 80, 100%	50, 80%
RCS and Secondary Chemistry and Radiochemistry Test	20, 50, 80, 100%	20, 50, 80, 100%
Turbine Trip Test	100%	100%
Unit Load Rejection Test	[redacted] 100%	[redacted] 100%
Shutdown from Outside the Control Room Test	≥ 10%	≥ 10%
Loss of Offsite Power Test	≥ 10%	≥ 10%
Biological Shield Survey Test	20, 50, 80, 100%	20, 50, 80, 100%
Xenon Oscillation Control Test	[redacted] ≥50%	N/A
Dropped CEA TEST	Post 80 [redacted]%	N/A
"Ejected" CEA Test	Post 80 [redacted]%	N/A
Steady-State Core Performance Test	20, 50, 80, 100%	20, 50, 80, 100%
Intercomparison of PPS, CPC and Process Computer Inputs	20, 50, 80, 100%	20, 50, 80, 100%
Verification of CPC Power Distribution Related Constants	20, 50% [redacted]	20, 50% [redacted]

* An Expanded test program is conducted for the "first-of-a-kind" in order to validate the design, the design methods, and the safety analysis assumptions.

** Reduced testing is contingent upon the demonstration that "Follow-On" plants behave in an identical manner as the "First-of-a-Kind" plant through conformance with the acceptance criteria given in Table 14.2-7.

*** Initial Power Level

**** temperature and The power coefficient measurements are done as close as possible to 100% power at a level where CEA motion is practical accounting for margin considerations

CESSAR

Table 14.2-7

PHYSICS (STEADY STATE) TEST ACCEPTANCE CRITERIA TOLERANCES

Parameter	First-of-a-kind	Follow-on Plant
LPTT		
Symmetry Test	± 1 1/2 ¢	± 1 1/2 ¢
CEA Group Worths	± 15% or .1% Δp whichever is greater	± 10% or .05% Δp whichever is greater
Total Worth (Not Shutdown)	±10%	±10%
Temperature Coefficient	± .5 x 10^{-4} Δp/°F	± .3 x 10^{-4} Δp/°F
Critical Boron Concentration	± 100 ppm	± 50 ppm
Boron Worth	± 15 ppm/% Δp	± 10 ppm/% Δp
Dropped and Ejected CEA WORTHS	±25% or .1% Δp whichever is greater	N/A
PAPT		
Power Distribution (Radial and Axial)	** RMS ≤ 5%	** RMS ≤ 3%*
Peaking Factors (Fxy, FR, Fz1, Fq)	± 10%	± 7.5%*
Temperature Coefficient	± .5 x 10^{-4} Δp/°F	± .3 x 10^{-4} Δp/°F
Power Coefficient	± .2 x 10^{-4} Δp/% power	± .2 x 10^{-4} Δp/% power
Pseudo Ejected CEA (2D Power Density RATIO Comparison)	± 20%	~~± 20%~~ N/A
Dropped CEA (2D Power Density RATIO Comparison)	± .2	~~N/A~~ N/A

* at 50% power and above

~~** ± 1¢ or 20% of the average pin power on a normalized distribution~~

** $RMS = \sqrt{\frac{\sum_{1}^{N} (RPD^{PRED} - RPD^{MEAS})^2}{N}}$

where N = 1, 2, 3 .... N (Total NUMBER OF FUEL ASSEMBLIES IN CORE OR NUMBER OF AXIAL PLANES, AS APPROPRIATE).

CESSAR AMENDMENT 10

EXCERPTS FROM DECEMBER 5, 1984

LETTER FROM A. E. SCHERER, CE, TO

D. G. EISENHUT, NRC (LD-84-070).

level, resulting in a transient pressure below normal operating pressure. To minimize the extent of this transient, the backup heaters are energized, contributing more heat to the water. Backup heaters are deenergized in the event of concurrent high-level error and high-pressurizer pressure signals. A low-low pressurizer water level signal deenergizes all heaters before they are uncovered to prevent heater damage. The pressure control program is shown in Figure 5.4.10-5.

5.4.10.3 Evaluation

It is demonstrated by analysis in accordance with requirements for ASME Code, Section III, Class 1 vessels that the pressurizer is adequate for all normal operating and transient conditions expected during the life of the facility. Following completion of fabrication, the pressurizer is subjected to the required ASME Code, Section III hydrostatic test and post-hydrostatic test non-destructive testing.

During hot functional testing, the transient performance of the pressurizer is checked by determining its normal heat losses and maximum ~~pressurization and~~ depressurization rates. This information is used in setting the pressure controllers.

Further assurance of the structural integrity of the pressurizer during plant life will be obtained from the inservice inspections performed in accordance with ASME Code, Section XI, and described in Section 5.2.

Overpressure protection of the Reactor Coolant System is provided by four ASME Code spring-loaded safety valves. Refer to Section 5.4.12 and 5.4.13.

5.4.10.4 Tests and Inspections

Prior to and during fabrication of the pressurizer, non-destructive testing is performed in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code. Table 5.4.10-2 summarizes the pressurizer inspection program, which also includes tests not required by the Code. Refer to Section 5.2.1 for inservice inspections of the pressurizer.

7.16.4.2 Provisions shall be made to preclude the introduction of air into the SCST ~~or shipping containers~~ during fill operations.

7.16.4.3 ~~If a pump is used to transfer hydrazine to the SCST, all of the metal components of the pump should be type 304 or 347 stainless steel. Seals should be the mechanical type and should be constructed according to the pump supplier's recommendations for hydrazine service.~~

INSERT (A)

7.16.5 FIRE PROTECTION

A fire protection system shall be provided to protect the Iodine Removal System and shall include, as a minimum, the following features:

a. Facilities for fire detection and alarming.

b. Facilities or methods to minimize the probability of fire and its associated effects.

c. Facilities for fire extinguishment.

d. Methods of fire prevention such as use of fire resistant and non-combustible materials whenever practical, and minimizing exposure of combustible materials to fire hazards.

e. Assurance that fire protection systems do not adversely affect the functional and structural integrity of safety related structures, systems, and components.

f. Care should be exercised to ensure fire protection systems are designed to assure that their rupture or inadvertent operation does not significantly impair the capability of safety related structures, systems, and components.

7.17 ENVIRONMENTAL

See Section 7.7 and CESSAR Section 3.11 for environmental interfaces.

7.18 MECHANICAL INTERACTION

7.18.1 IRS components shall be properly supported such that pipe stresses and support reactions are within allowable limits, as defined in CESSAR Section 3.9.2. CE provides the Applicant the loads at the supports/structures interface locations for components that CE supplies, under normal, upset, emergency, faulted, and test conditions, as described in CESSAR Section 3.8.5.

7.18.2 IRS piping and fittings shall be Seismic Category I.

INSERT (A).

7.16.4.3 All transfer lines and pump components in contact with the hydrazine solution should be clean and hydrazine compatable as recommended by the chemical manufacturers.

1.75, "Physical Independence of Electric Systems". A discussion of the physical independence is provided below which describes the compliance with Section 4.6 of IEEE 279-1971 and General Design Criteria 3 and 21. General Design Criterion 17 is discussed in the Applicant's Safety Analysis Report.

The PPS cabinet is divided into four bays which are separated by mechanical and thermal barriers. Each bay contains one of the four redundant channels of the RPS and ESFAS. This provides the separation and independence necessary to meet the requirements of Section 4.6 of IEEE 279-1971.

INSERT

~~Tests or analyses will be performed to demonstrate that no single credible event in one PPS bay can propagate the fault created to any other bay. The tests or analyses will be completed about two months prior to the fabrication of the equipment for installation in the first Applicant's facility and will be submitted for review prior to installation.~~

The ESFAS Auxiliary Relay Cabinets provide separation and independence for the selective two-out-of-four actuation logics and actuation relays of the two redundant ESF Systems' Trains. Each train's logic and relays are contained in a separate cabinet with all of the train A actuation circuits in one cabinet and all of the train B actuation circuits in the other cabinet. There are mechanical and thermal barriers within the cabinets to protect different portions of the selective two-out-of-four logic from spurious actuation. The two cabinets are physically separated from each other.

The RTSS consists of four RTSG. Each RTSG and its associated switches, contacts, relays, etc. is contained in a separate cabinet. Each cabinet is physically separated from the other cabinets. This method of construction ensures that a single credible failure in one RTSG cannot cause malfunction or failure in another cabinet.

The separation and independence of the power supplies for each of the above systems is discussed in Chapter 8.0. The interface requirements appear in Section 7.1.3 while the implementation will appear in the Applicant's Safety Analysis Report. Protection system analog signals, sent to the Plant Monitoring System (PMS), are isolated from the protection system. Digital signals are also isolated for the associated signals coming from the protection system.

All of these isolation techniques ensure that no credible failures on the output side of the isolation device will effect the PPS side and that the independence of the PPS is not jeopardized. The test results reports on the isolation devices (within CESSAR Licensing scope) will be submitted for review prior to installation of the devices in the first Applicant's facility.

7.1.2.11 Conformance to IEEE 387-1972

Conformance to IEEE 387-1972 , "IEEE Trial-Use Standard: Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations", as criteria in the design of these systems is discussed in the Applicant's Safety Analysis Report.

CESSAR 7.1.2.10

Replace paragraph 3 with the following paragraph.

Separation of redundant Class 1E circuits within the PPS cabinet is accomplished through 6 inch separation or barriers or conduit. However, in the formation of the logic matrices (AB, AC, BC, AD, BD, CD), initiation circuits, and actuation circuits, 6 inch separation is not maintained, nor can barriers or conduit be utilized. An analysis has been performed to show that the separation achieved is acceptable. Tests and analyses have also been completed to demonstrate that no single credible event in one PPS bay can prevent the circuitry in any other bay from performing its safety function.

TABLE 9.2-1

~~PRIMARY AND SECONDARY~~ SYSTEM DEMINERALIZER EFFLUENT
MAKEUP WATER LIMITS

pH*	6.0 to 8.0
Conductivity	Less than ~~1~~ 0.2 µmhos
Chloride	Less than ~~0.15~~ 0.005 ppm Cl
Fluoride	Less than ~~0.10~~ 0.005 ppm F
~~Suspended Solids~~	~~Less than 0.5 ppm~~
~~Gaseous**~~	~~Non-Deaerated/Deaerated~~
Silica ~~(SiO_2)~~ ($Si O_4$)	Less than 0.01 ppm
Sodium	Less than 0.003 ppm

* If water contains CO_2, the pH specification may be lowered to 5.8 to compensate for CO_2 absorption.

~~** Deaeration gives conservatism to makeup water system design but it is not considered necessary.~~

~~*** Pertains to secondary makeup water only.~~

time when exposed to normal reactor coolant chemistry conditions, approaching low steady state values within approximately 200 days. The high pH condition produced by high ammonia concentration (to 50 ppm) minimizes corrosion product release and assists in the rapid development of the passive oxide film. Most of the film is established within 7 days at hot, high pH conditions.

To aid in maintaining the pH during this passivation period, lithium in the form of lithium hydroxide, is added to the coolant and maintained within a 1-2 ppm lithium-7 range.

At power, oxygen concentration is limited by maintaining excess dissolved hydrogen gas in the coolant. The excess hydrogen forces the water decomposition/synthesis reaction in the reactor core to water rather than hydrogen and oxygen. Oxygen in the makeup water is removed in the same way.

In order to minimize the effect of crud deposition on the reactor core heat transfer surfaces, lithium-7 hydroxide additions to the reactor coolant are made. The lithium-7 hydroxide produces pH conditions within the reactor coolant at operating temperature which reduces the corrosion product solubility and, hence, the dissolved crud inventory in the circulating reactor coolant. The elevated pH promotes conditions within the coolant for selective deposition of corrosion products on cooler surfaces (SG) rather than hotter surfaces (core). An additional advantage is the formation of a more stable and tenacious passive oxide layer on out-of-core system surfaces. The lithium concentration is maintained within a ~~0.2-1.0~~ 1.0-2.0 ppm lithium-7 range during operation. ~~Late in core life, when boron concentration is low, maintaining lithium at the high end of this band could lead to the formation of lithium metaborates. For this reason, when the deborating ion exchanger is placed on line for boron control (30 ppm boron) the lithium concentration is decreased and maintained within 0.2-0.6 ppm lithium. Lithium is not controlled (i.e., there is no minimum concentration) during refueling operations.~~

9.3.4.1.3.3 Reactivity Control. Boron concentration is normally controlled by feed and bleed. To change concentration, the makeup system supplies either reactor makeup water or boric acid to the Volume Control Tank, and the letdown stream is diverted to the holdup tanks via the preholdup ion exchanger and the gas stripper. Toward the end of a fuel cycle, with low boric acid concentration in the coolant, feed and bleed becomes inefficient, and the deborating ion exchanger is used to reduce the RCS boron concentration. The ion exchanger contains an anion resin initially in the hydroxyl form, which is converted to a borate form as boron is removed from the reactor coolant.

9.3.4.2 System Description

9.3.4.2.1 System

The normal reactor coolant flow path through the CVCS is indicated by the heavy lines on the Piping and Instrumentation Diagrams, Figures 9.3-1 through 9.3-4.

TABLE NO. 9.3-1

(Sheet 1 of 2)

OPERATING LIMITS

REACTOR COOLANT

1.0 MAKEUP WATER

Analysis	Normal	~~Abnormal~~
Chloride (Cl)	<0.15 ppm	~~< .15 ppm~~
~~Silica (SiO_2)~~	~~<0.01 ppm~~	~~< .02 ppm~~
~~Conductivity~~	~~<1.0 μmhos/cm~~	~~< 2 μmhos/cm~~
pH	6.0 - 8.0 (1)	-
Fluoride (F) ~~(2)~~	<0.1 ppm	~~<0.1 ppm~~
Suspended Solids	<0.5 ppm	-

2.0 PRIMARY WATER

Analysis	Pre Core Hot Functionals ~~(3)~~(2)	Initial Core Load and ~~Initial~~ Criticality	~~Power~~ Operation
pH (77°)	9.0 - 10.4	4.5 - 10.2 ~~(4)~~	4.5 - 10.2 ~~(4)~~
Conductivity	~~(5)~~ (3)	~~(5)~~ (3)	~~(5)~~ (3)
Hydrazine	30 - 50 ppm (4)	30 - 50 ppm (4)	1.5 x Oxygen ppm (5) (max. 20 ppm)
Ammonia	<50 ppm	<50 ppm	<0.5 ppm
Dissolved Gas	---	---	(6)
Lithium	1-2 ppm	1.0 - 2.0 ~~0.2 - 1.0~~ ppm ~~(7)~~	1.0 - 2.0 ~~0.2 - 1.0~~ ppm ~~(8)~~
Hydrogen	---	--- (7)	25 ~~10~~ - 50 cc (STP)/kg (H_2O) (8)
Oxygen	≤0.1 ppm	≤0.1 ppm ~~(9)~~(10)	≤0.1 ppm (10)
Suspended Solids	<0.5 ppm, 2 ppm max. (9)	<0.5 ppm, 2 ppm max. (9)	<0.5 ppm, 2 ppm max. (9)
Chloride	≤0 15 ppm	≤0.15 ppm	≤0.15 ppm
Fluoride	≤0.1 ppm	≤0.1 ppm	≤0.1 ppm
Boron	---	<Refueling Concentration	~~<4400 ppm~~ ≤ Refueling Concentration

Notes for Table No. 9.3-1

(1) May be as low as 5.8 if proven due to CO_2 absorption.

~~(2) Fluoride limit applicable to water intended for primary makeup.~~

(2) ~~(3)~~ Special hot conditioning limits:

Temperature >350°F for 7-10 days

~~(4) During core load and refueling, pH may be as low as 3.8 due to high boron concentration.~~

(3) ~~(5)~~ Consistent with additive concentration.

INSERT (A)

(6) Prior to a depressurization shutdown, reduce total gas to <10cc(STP)/kg (H_2O) to limit the possibility for explosive mixtures.

INSERT (B)

~~(7) If the purification ion exchanger is being saturated with ^7Li on line, maintain 1-2 ppm ^7Li until saturation is reached (indicated by ^7Li breakthrough), then revert to the 0.2-1 ppm range. This range must be in force prior to criticality. The lithium range does not apply during core loading.~~

~~(8) Near end of life, when the deborating ion exchanger is placed in service (~20 ppmB) the lithium limit is 0.2-0.5 ppm.~~

~~(9) Not applicable during core load.~~

INSERT (A)

(4) Hydrazine is maintained at 30-50 ppm any time the RCS is less than 150°F.

(5) Prior to exceeding 150°F during heatup or below 400°F during cooldown.

INSERT (B)

(7) During the transition from post-core to operating, hydrogen should be maintained in the 15 to 25 cc(STP)/kg(H range to minimize degassing requirements in case the reactor plant must be shutdown and depressurized.

(8) Hydrogen should be <5 cc(STP)/kg(H_2O) before securing the reactor coolant pumps.

(9) The abnormal condition of 0.5 to 2.0 ppm is permitted for up to 14 hours to allow for crud burst conditions.

(10) Not applicable during core load.

Secondary water chemistry is based on the zero solids treatment method. This method employs the use of volatile additives to maintain system pH and to scavenge dissolved oxygen present in the feedwater.

A neutralizing amine is added to establish and maintain alkaline conditions in the feedtrain. Neutralizing amines which can be used for pH control are ammoniz, morpholine, and cyclohexylamine. Ammonia should be used in plants employing condensate polishing to avoid resin fouling. Although the amines are volatile and will not concentrate in the steam generator, they will reach an equilibrium level which will establish an alkaline condition in the steam generator.

Hydrazine is added to scavenge dissolved oxygen present in the feedwater. Hydrazine also tends to promote the formation of a protective oxide layer on metal surfaces by keeping these layers in a reduced chemical state.

Both the pH agent and hydrazine can be injected continuously at the discharge headers of the condensate pumps or condensate demineralizer, if installed. These chemicals are added as necessary for chemistry control, and can also be added to the upper steam generator feed line when necessary.

Operating chemistry limits for ~~feedwater and~~ secondary steam generator ~~water~~ water and feedwater and condensate are give in Tables 10.3.4-1 and 10.3.4-2.

The limits stated are divided into two groups; normal and abnormal. The limits provide high quality chemistry control and yet permit operating flexibility. The normal chemistry conditions can be maintained by any plant operating with little or no condenser leakage. The abnormal steam generator limits are suggested to permit operations with minor system fault conditions until the affected component can be isolated and/or repaired.

The following procedures are recommended to the applicant:

~~If the continuous monitors on the steam generator blowdown sample lines detect more than four μmhos/cm specific conductivity, the steam generator water should be immediately sampled and analyzed for chloride concentration:~~

~~A. If chloride concentration is within baseline* values, operations may continue. The generators should be sampled for chlorides at least once per eight hour shift while conductivity exceeds four μmhos/cm.~~

~~B. If chloride concentration is in excess of baseline* values or approaching 0.1 ppm, a condenser leak is indicated and leak isolation procedures should be instituted.~~

~~*Baseline values are defined as the chloride concentration which exists in the steam generator water at steady state operating conditions without condenser leakage. A typical baseline chloride concentration is expected to be approximately 20 ppb.~~

INSERT (A)

INSERT (A)

When the normal range is exceeded, immediate investigation of the problem should be initiated, sampling frequency increased to the abnormal level (at least twice per 8 hour shift) and blowdown increased to one (1) percent of the main steaming rate. The problem should be corrected and the parameter(s) returned to the normal range within one week. If this cannot be done, and the parameter has a listed abnormal range, power should be reduced to 25% as if the abnormal range had been exceeded.

When the abnormal range is exceeded, power should be reduced to the lowest value (maximum of 25%) consistent with automatic operation of the feed system. Continued plant operation is then possible while corrective action is taken. Power reduction should be initiated within four hours of exceeding the abnormal range. The problem should be corrected and the parameter(s) returned to the normal range within one hundred (100) hours. If this cannot be done, the unit should be shutdown.

~~If any of the normal operating specifications in Table 10.3.4-1 are exceeded, immediate investigation of the problem should be initiated, sampling frequency increased to the abnormal level (at least twice per eight hour shift) and blowdown increased to one percent of the main steaming rate. If condenser leakage is indicated, leak isolation procedures should be instituted. Upon verification of a condenser leak, with steam generator chloride in excess of 0.1 ppm, a plant power reduction should be initiated immediately and should be reduced to 25% within four hours. Power should not exceed 25% until chloride concentration is within the normal specifications.~~

~~In the event condenser leakage causes any operating specifications in Table 10.3.4-1 to equal or exceed abnormal limits, a shutdown should be completed within four hours.~~ Draining or flushing of the steam generators will be necessary to reduce the impurity concentration.

10.3.4.2 Corrosion Control Effectiveness

Alkaline conditions in the feedtrain and the steam generator reduce general corrosion at elevated temperatures and tend to decrease the release of soluble corrosion products from metal surfaces. These conditions promote the formation of a protective metal oxide film and thus reduce the corrosion products released into the steam generator.

Hydrazine also promotes the formation of a metal oxide film by the reduction of ferric oxide to magnetite. Ferric oxide may be loosened from the metal surfaces and be transported by the feedwater. Magnetite, however, provides an adherent protective layer on carbon steel surfaces. Hydrazine also promotes the formation of protective metal oxide layers on copper surfaces.

The removal of oxygen from the secondary waters is also essential in reducing corrosion. Oxygen dissolved in water causes general corrosion that can result in pitting of ferrous metals, particularly carbon steel. Oxygen is removed from the steam cycle condensate in the main condenser deaerating section. Additional oxygen protection is obtained by chemical injection of hydrazine into the condensate stream. Maintaining a residual level of hydrazine in the feedwater ensures that any dissolved oxygen not removed by the main condenser is scavenged before it can enter the steam generator.

The presence of free hydroxide (OH^-) can cause rapid corrosion (caustic stress corrosion) if it is allowed to concentrate in a local area. Free hydroxide is avoided by maintaining proper pH control, and by minimizing impurity ingress in the steam generator.

TABLE 10.3.4-1

OPERATING CHEMISTRY LIMITS FOR SECONDARY STEAM GENERATOR WATER

Variable	Normal (1) Specifications	Abnormal ~~(2)~~ Limits
pH (mixed system)(2)	8.5 ~~8.2~~ - 9.2	~~7.5 8.2~~
(copper free)	9.0 - 9.6	~~9.2 9.3~~
Cation ~~Specific~~ Conductivity (3)	0.8 < ~~7~~ µmhos/cm ~~(3)~~	0.8 - 2.0 ~~7 15~~ µmhos/cm
~~Suspended Solids~~	~~< 1.0 ppm~~	~~1 - 10 ppm~~
Silica	<300 ppb ~~< 1.0 ppm~~	~~1 10 ppm~~
Chloride	<20 ppb ~~< 0.1 ppm~~	20 - 100 ppb
Sodium (4)	< 20 ppb	20 - 100 ppb
Sulfate	< 15 ppb	15 - 100 ppb

NOTES:

(1) Normal specifications are those which should be maintained by continuous steam generator blowdown during proper operation of secondary systems.

~~(2) Abnormal limits indicate a fault condition exists and plant shutdown should be commenced if abnormal limits are exceeded for 4 hours.~~

~~(3) Specific conductivity alarm set point: 4 µmhos/cm.~~

(2) Mixed system is any secondary system containing copper alloy components.

(3) If the immediate shutdown limit of 7.0 µmhos/cm is exceeded the unit should be shutdown within four hours.

(4) If the immediate shutdown limit of 500 ppb is exceeded the unit should be shutdown within four hours.

TABLE 10.3.4-2

OPERATING CHEMISTRY LIMITS FOR FEEDWATER AND CONDENSATE

Variable	Normal[1] Specifications	~~Abnormal[2] Limits~~
pH		
a. Mixed ~~Feedwater~~ system ~~containing copper-alloy components~~	8.8 - 9.2	
b. Copper-free ~~feedwater~~ system	9.3 - 9.6 ~~9.2 9.6~~	
Conductivity (Intensified cation) (Feedwater)	< 0.2 ~~0.5~~ µmhos/cm	~~0.6 1.6 µmhos/cm~~
Hydrazine (Feedwater)	10 - 50 ppb	
Dissolved Oxygen		
(Feed)	< 3 ~~5~~ ppb	~~5 10 ppb~~
(Condensate)	< 10 ppb[2]	~~10 ppb~~
Sodium[3]	< 3 ~~1~~ ppb	~~1 10 ppb~~
Copper (Feedwater)	~~(3)~~ < 2 ppb[4]	~~(3)~~
Iron (Feedwater)	< 20 ~~10~~ ppb	~~10 30 ppb~~
~~Ammonia~~		
~~(with copper)~~	~~< 1.0 ppm~~	
~~(without copper)~~	~~< 2.0 ppm~~	
~~Silica~~	~~< 10 ppb~~	
pH Control Additive	(5)	

NOTES:

(1) Normal specifications are those which should be maintained during proper operation of secondary systems.

~~(2) Abnormal limits indicate a fault conditions exists and plant shutdown should be commenced if abnormal limits are exceeded for 4 hours.~~

~~(3) Eliminate copper ingress to the steam generators where practical.~~

(2) The condensate abnormal limit is 10-30ppb but the requirement for immediate shutdown does not apply even if the problem is not corrected within 100 hrs.

(3) For the condensate, sodium is monitored at each condenser hot well.

(4) Analysis not required for copper free systems.

(5) Limit is dependent upon pH.