



October 27, 1993
LD-93-153

Docket No. 52-002

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: System 80+TM Information for Issue Closure

Dear Sirs:

The attachments to this letter provide material to close follow-on questions to DSER responses. Attachment 1 provides the report on the System 80+ hydrogen mitigation system, which should be given to Mr. M. Snoderly. Attachment 2 provides markups of Sections 3.1 and 3.2 of the technical specifications. These should be given to Ms. A. Chu. Attachment 3 transmits revisions to Section 9.1 to resolve staff questions and to reflect minor design changes. Attachment 4 transmits copies of three faxes to Mr. R. Palla and Attachment 5 provides five faxes to Mr. N. Saitos.

If you have any questions, please call me or Mr. Stan Ritterbusch at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

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ATTACHMENT 1

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APPENDIX 19.11K

ASSESSMENT OF THE SYSTEM 80+ HYDROGEN
MITIGATION SYSTEM FOR APPLICATION IN A
SEVERE ACCIDENT ENVIRONMENT

PREPARED BY

ABB-COMBUSTION ENGINEERING
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OCTOBER 1993

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TABLE OF CONTENTS

<u>SECTION</u>	<u>TITLE</u>	<u>PAGE NO.</u>
1.0	INTRODUCTION	K-1
2.0	BACKGROUND	K-1
3.0	SYSTEM 80+ HYDROGEN CONCENTRATION	K-1
4.0	OVERVIEW OF EXPERIMENTAL RESEARCH RELATED TO HYDROGEN COMBUSTION AND CONTROL	K-2
4.1	SUMMARY OF RESEARCH ON LEAN FLAMMABILITY LIMIT FOR DETONATION	K-2
4.2	REVIEW OF HYDROGEN MIXING AND DISTRIBUTION EXPERIMENTS	K-6
4.3	EXPERIMENTAL EVIDENCE SUPPORTING CONTROL OF HYDROGEN VIA DELIBERATE IGNITION SYSTEMS	K-9
5.0	HYDROGEN IGNITER PLACEMENT GUIDELINES FOR SYSTEM 80+	K-23
5.1	HMS DESIGN GOALS	K-23
5.2	HMS PLACEMENT CRITERIA	K-23
6.0	DESCRIPTION OF THE SYSTEM 80+ HYDROGEN MITIGATION SYSTEM	K-32
6.1	IGNITER PLACEMENT	K-32
6.2	IGNITER POWER SUPPLY	K-32
6.3	IRWST VENT SYSTEM	K-32
6.4	IGNITER ACCESSIBILITY AND MAINTENANCE	K-34
7.0	ANALYTICAL VERIFICATION OF THE SYSTEM 80+ HYDROGEN IGNITER SYSTEM	K-46
7.1	CONSTRUCTION OF PLANT MODEL	K-46
7.2	METHODOLOGY	K-47
7.3	RESULTS OF CALCULATIONS	K-48
7.4	CONCLUSIONS	K-51

DRAFT.

TABLE OF CONTENTS

<u>SECTION</u>	<u>TITLE</u>	<u>PAGE NO.</u>
8.0	ASSESSMENT OF THE SYSTEM 80+ HYDROGEN DETONATION ISSUES	K-55
8.1	LIKELIHOOD OF DETONATION AND CONTAINMENT SURVIVABILITY	K-55
8.2	COMMENTS ON THE POTENTIAL FOR A CONDENSATION INDUCED DETONATION	K-58
9.0	REFERENCES	K-59

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1.0 INTRODUCTION

The purpose of this appendix is to provide details with regard to the purpose, design, implementation and capabilities of the System 80+ Hydrogen Mitigation System (HMS) in responding to a severe accident. Additional detail on the HMS can also be found in Section 6.2.5.

2.0 BACKGROUND

The accident at TMI indicated that severe accidents can release large quantities of hydrogen to containment. As it was further demonstrated, this hydrogen can accumulate and undergo combustion potentially threatening both the survivability of safety equipment and containment integrity. As a consequence of these observations the NRC identified beyond design basis hydrogen control as an unresolved safety issue (USIA-48). This issue covers hydrogen control measures for recoverable degraded-core accidents for all Mark I, II and III boiling water reactors and pressurized water reactors with an ice condenser containment. At that time PWRs with large dry containments were excluded from USIA-48 and the issue for large dry PWRs was investigated as Generic Issue 121 (GI-121). USIA-48 was resolved by an amendment to the 10CFR50.441, "Hydrogen Control Systems", which required the subject reactor to implement a hydrogen control system capable of "accommodating an amount of hydrogen equivalent to that generated from the reaction of 75 % of the fuel cladding with water, without loss of containment integrity." Furthermore, the NRC reaffirmed its policy that the "prevention of excessive radiation doses to the public can best be assured by maintaining a leak tight containment and that this, in turn, can be provided by assuring that there is structural integrity with margin" and that sufficient equipment will be available to establish and maintain safe shutdown. At that time, the regulation was based on the lower limit for hydrogen detonation to initiate at a hydrogen concentration of 13 volume per-cent. If it could be demonstrated that the hydrogen concentration for this level of oxidation would remain below 13 volume percent no active system would be required to prevent detonations. For future plants, this issue was folded into the Containment Performance Improvement (CPI) Program and was defined in SECY-88-147², SECY-90-016³ and SECY-93-087⁴. These features were incorporated into the Code of Federal Regulations as Post TMI rule 10CFR50.34(f)⁵. In defining this regulation for advanced LWRs the hydrogen control requirement was further tightened such that the plants would be able to accommodate the amount of hydrogen equivalent to that generated from the reaction of 100 % of the fuel active cladding with water and maintain the hydrogen concentration in containment to below 10 volume percent hydrogen.

3.0 SYSTEM 80+ HYDROGEN CONCENTRATION

The System 80+ containment structure was sized to accommodate the hydrogen requirements of the EPRI Utility Requirements Document⁶. These requirements were consistent with the guidance in 10CFR50.4. Consequently, the System 80+ containment was sized to over 3 million cubic feet. Scoping calculations of the hydrogen concentration resulting from a 75 % oxidation of the fuel cladding resulted in a containment volumetric

hydrogen concentration of 10.2 %. In the context of the previous regulation, System 80+, by design, would have passively eliminated the hydrogen detonation threat. Several years later the hydrogen detonation threat was redefined. The upper limit of oxidation to be accommodated by the containment increased from 75 to 100 % of active cladding while simultaneously, the minimum detonation limit dropped to 10 volume percent. As a result of this redefinition of the hydrogen source, the System 80+ containment volumetric hydrogen concentration increased to 13.6 percent. Thus, in order to meet the requirements set forth in 10CFR50.34(f) an active means of hydrogen control is required for the System 80+ design. An effective and cost-efficient method for achieving this goal was by the addition of a hydrogen igniter system.

4.0 OVERVIEW OF EXPERIMENTAL RESEARCH RELATED TO HYDROGEN COMBUSTION AND CONTROL

This section provides an overview summary of the key experimental results that were used in guiding the development of a hydrogen igniter system for System 80+. Since the existing System 80+ containment design is sufficiently robust to withstand containment threats associated with deflagrations (see Section 19.11), the primary goal of the deliberate ignition system required by 10CFR50.34(f) would be to preclude the potential for a detonation. Therefore, this review has concentrated on experiments that provided information on the following:

1. limits on detonability and the likelihood of detonation
2. hydrogen mixing and distribution
3. effectiveness and performance of deliberate ignition systems in hydrogen control.

This information was used to help define igniter system design criteria and performance goals for the System 80+ hydrogen igniter system design. These issues are investigated in detail below.

4.1 SUMMARY OF RESEARCH ON THE LEAN FLAMMABILITY LIMIT FOR DETONATION

The purpose of the hydrogen igniter system for the System 80+ ALWR is to provide a means of hydrogen control so that a hydrogen detonation within the System 80+ containment could be averted even following the very low probability severe core damage events. To place the purpose of this system in perspective it is worthwhile to briefly consider the current state of the art in our understanding of hydrogen detonation limits.

Studies regarding the estimation of the lower detonability limit for hydrogen-air and hydrogen-air-steam mixtures have not as yet completely defined this function. The history of detonability research spans many decades. There are two ways in which a detonation can occur in a detonable mixture. One is direct ignition and the other is flame acceleration. Based on estimates of the energy needed to ignite a detonable mixture at 13 volume percent hydrogen, the National Academy of Sciences (NAS) noted that direct ignition detonation within the

containment is unlikely¹⁰.

The other means by which a detonation can occur is flame initiation and acceleration. Flame acceleration can occur due to turbulence, changes in geometry, obstacles and wall roughness. This process is termed Deflagration-to-Detonation Transition (DDT).

As late as the 1950s the lower detonability limit of a hydrogen-air mixture was estimated to be at 18 volume percent hydrogen. However, much of the experimentation used to support this value were small scale and employed relatively uncomplicated geometries (i.e., spheres, circular tubes, etc.). As larger facilities were used for experimentation and a wider range of geometries tested, evidence began to develop that suggested that the actual detonability limit is lower than the 18 volume per cent previously reported by Elbe⁷ and others. Furthermore, the importance of geometric features (size, obstacles, vents etc.) on detonation began to become more apparent.

In the mid-1980's, in an effort to investigate the role of flame acceleration on producing detonations in reactor geometries, DDT tests were conducted by Sandia National Laboratory in the FLAME 31 and MINIFLAME 32 facilities. The FLAME facility consists of a 1:2 scale model of the upper plenum volume of a PWR ice condenser containment. These tests investigated DDT for hydrogen concentrations between 12 and 30 percent and included several parametric studies regarding the importance of obstacles and transverse venting on DDT. For all geometries tested no significant flame acceleration was noted at hydrogen concentrations of 12%. DDT was first observed at 15% hydrogen for tests with obstacles present and no transverse venting. At the 25 to 30% hydrogen concentration, DDT was observed without the presence of obstacles. Smaller scale experiments were performed at the MINIFLAME facility. Because of the smaller size, DDT was not observed at hydrogen concentrations of 20%.

Recently, with a carefully configured geometry Dorofeev¹¹ succeeded in achieving a Deflagration Detonation Transition (DDT) at a hydrogen concentration of only 12.5 volume percent.

Much of this experimental evidence has been reviewed and evaluated. An independent review of the hydrogen detonation issue by the National Academy of Sciences concluded that a hydrogen concentration of 13 volume percent is a reasonable lower limit for expected hydrogen detonability in a PWR containment atmosphere with small quantities of steam¹⁰. More recently, Shepard, as a contractor for NRC, concluded that a conservative lower bound for the hydrogen concentration would be 10 volume percent¹².

As can be seen from the above experimentation the lower hydrogen detonation limit appear to be conservatively bounded by about 10 volume percent hydrogen. This lower limit then replaced the previous 13 volume percent considered in evaluating existing igniter systems for operating plants. From its inception, the System 80+ design philosophy was to overwhelm a potential problem by design. To this end it was a goal in the System 80+ design to demonstrate that for a large amount of core damage (equivalent to 75% zircaloy oxidation), the hydrogen concentration would not be in the detonable range. In this effort a lower bound global hydrogen concentration of 10 volume percent was achieved. This desire

factored in part to the large 3.4 million cubic feet System 80+ containment design. Thus, for all accident scenarios with a core melt probability above 10^{-6} , detonation is indeed passively precluded by judicious design. The purpose of the ignition system then is to meet regulatory guidance which to a large extent requires the ability of System 80+ to provide active systems to preclude a detonation in what would otherwise be expected to be an unrecoverable core melt sequence. These events have a cumulative occurrence frequency of less than 10^{-6} per year.

Application to PWRs

One important outcome of the Sandia FLAME experiments was the development of the Sherman/Berman qualitative detonability likelihood criteria^{9,13}. In this system, the authors rated the detonation potential on a 5 point scale, with 1 being most detonable and 5 being virtually undetonable. The mixture detonability was based on two parameters: (1) the detonation cell width (which is directly related to hydrogen concentration), and (2) physical plant geometry. A review of all internal compartments for System 80+ suggests that System 80+ is a class 4 containment. This rating implies containment conditions are not conducive to DDT and that the potential for a detonation is unlikely to impossible. For completeness a summary of the Sherman/Berman geometrical ratings for System 80+ is presented in Table 4.1-1.

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TABLE 4.1-1
SHERMAN/BERMAN RANKING

Region	Description	Rank
1	containment upper dome	4,5
2	volume inside crane wall and above refueling pool	3,4
3	annular region outside crane wall above 115 ft elevation	3,4
4	HVAC distribution header	3
5	reactor cavity	3
6	cavity ventilation room	3,4
7	reactor cavity annular gap	2
8	refueling pool	4
9	steam generator compartments	2,3
10	pressurizer compartment	2
11	holdup volume	3
12	letdown heat exchanger room	3
13	regenerative heat exchanger room	3
14	volume inside crane wall between 91-9 and 115-6 elevations	3

Sherman/Berman Ranking Criteria

- Class 1: Large partially confined geometry with obstacles in the path of expanding unburned gases (geometry typically closed at one end).
- Class 2: Similar geometry to class 1, but "room" may be open at both ends or transverse venting is available
- Class 3: Open regions without obstacles
- Class 4: Large volumes with few obstacles and significant venting
- Class 5: Unconfined geometry

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4.2 REVIEW OF HYDROGEN MIXING AND DISTRIBUTION EXPERIMENTS

The ability of the hydrogen generated at the source to mix throughout the containment is important to ensure that locally high hydrogen concentrations will not develop. Overall results of hydrogen mixing experiments suggest a propensity for good mixing within various containment volumes, however mixing was found to be dependent on the relative location of the source hydrogen to the volume under concern. On microscopic levels, mixing at the point of the source is still expected to produce very localized large concentration gradients. This section summarizes some of the highlight information on hydrogen mixing.

The primary processes which govern the mixing of gaseous mixtures are forced and natural convection. The release of hydrogen and steam from the reactor system in the form of a jet flow would cause forced convective mixing. Buoyancy forces will induce natural convecting flows. During an accident it is expected that mixing will be promoted by a combination of forced and natural convection. The degree of mixing is dependent upon the hydrogen/steam release rate, fluid movement and turbulence introduced due to the flow.

A summary of hydrogen mixing tests are provided below.

Hanford Engineering Development Laboratory (HEDL) Mixing Tests¹⁴

Tests were performed to establish the hydrogen-steam/ helium-steam mixing capability within the lower compartment of the ice condenser. The test facility consists of a 20 m high, 7.6 m diameter vessel. While results of these tests are not directly applicable to large dry PWRs there does exist some level of commonality. Key observations from the HEDL program are summarized below.

1. The compartment was well mixed during the source release period with the maximum helium or hydrogen concentration differences of about 3 volume percent between points in the test compartment. This results in a peak concentration to average concentration ratio of less than 1.15.
2. Gas entrainment due to the high velocity jet was the dominant mixing process during the release period. Mixing levels were independent of the orientation of the source jet.
3. After termination of the source, the containment mixing was supported by the natural circulation process.

Nevada Test Site (NTS) Continuous Injection Tests¹⁵

The Nevada Test Site facility was used to study hydrogen combustion and mixing phenomena in a large scale (2100 m³, 75,000 ft³) spherical shell. As part of this test program continuous injection experiments were included to study the hydrogen mixing process in the presence and absence of combustion.

A. Hydrogen Mixing in the Absence of Combustion

At low pure hydrogen injection rates (about 0.4 kg/minute) and a quiescent atmosphere in the NTS facility was observed to fill with hydrogen from the top down. The location of the source was important to the gross hydrogen mixing process. In the absence of mixing fans, the volume located above the hydrogen source was observed to have a hydrogen concentration twice that of the volume below the source. The mixing process was markedly improved by the operation of water sprays and uniform vessel conditions were established "immediately".

Additional NTS experiments demonstrated that even in initially quiescent atmospheres, the injection of large quantities of steam along with the hydrogen (even at low hydrogen injection rates) would rapidly produce a well mixed mixture within the facility.

B. Hydrogen Mixing During Continuous Combustion

In this experiment, hydrogen and steam injection rates were 1.9 kg/min and 30 Kg/min respectively and the atmosphere was initially quiescent with a steam concentration of 30 %. The injection process caused a rapid dispersion of hydrogen into the containment upper dome. Ignition occurred at the top of the facility. Shortly thereafter, the flame had become attached to the hydrogen source. At that point incoming hydrogen was efficiently consumed and global hydrogen concentrations were reduced.

CEA-SACLAY Hydrogen Simulant Mixing Experiment¹⁶

A series of light gas mixing experiments were performed by the Commissariat a l'Energie Atomique in Saclay, France. The purpose of this investigation was to estimate the potential for hydrogen stratification within the containment following a severe accident. These experiments were conducted on a small scale test apparatus with a volume of 240 ft³ and a height of 7.5 ft. Helium was selected as a simulant mixing gas. The test conditions and facility were scaled according to prototype Froude and Reynolds numbers. The Reynolds number for the injected flow varied from 300 to 10,000 and bounded typical System 80+ hydrogen release rates.

All experiments were conducted with the helium source located at the floor of the facility. The facility was instrumented to monitor the gas concentration axially throughout the scaled containment. The transitory mixing process was then measured as a function of several driving parameters including the source injection Reynolds number. These results suggest that the hydrogen mixing process is very effective over a wide range of Reynolds numbers. Asymptotic concentration gradients resulted in maximum-to-average concentration ratios of typically less than 1.07. Transitory mixing was also shown to approach an equilibrium quickly with maximum-to-average concentration ratios in the vicinity of 1.15 ten minutes after the cessation of the injection.

HDR Hydrogen Mixing Experiments^{17,32}

The HDR (Figure 4.2-1) is a decommissioned reactor facility in Germany. Over the past decade this facility has been used for a wide variety of large scale reactor experiments. Recently, the HDR has been utilized in

the investigation of hydrogen mixing phenomena. Unlike the experiments described above which were single volume tests, the HDR containment is a complex building with 72 sub-compartments with over 300 interconnecting flowpaths. Consequently, this facility was intended to provide experimental data on the long term gas transport behavior in a large scale, multi-compartment facility in the presence of steam under natural convection conditions. The total volume of HDR is 11,300 m³ with a height of 60 m. Several HDR experiments were performed. For purposes of safety, the hydrogen gas injection was simulated by a 85/15 helium/hydrogen mixture. While all the experiments are not readily available for review several findings from the experiments are worthy of note.

A. Importance of Injection Location

The HDR tests E11.2 and E11.4 illustrated the importance of injection location on hydrogen mixing and stratification. Both experiments consisted of small break LOCAs with a delayed hydrogen release. In test E11.2 the hydrogen and steam sources are located at the 23 m elevation, while for test E11.4 the sources were located much lower in the containment at the +2 m elevation. Results of test E11.2 indicated that the hydrogen distributed itself into two regions. Below the gas source the hydrogen simulant ("gas") concentration remained very low (below 5 volume percent) while above the source "gas" concentrations exceeded 20 volume percent (See Figure 4.2-2). Once this stratification was setup the HDR atmosphere could not be homogenized by operational measures. In test E11.4, the low position of the gas release resulted in good mixing of the hydrogen (See Figure 4.2-3) with typical local concentration gradients of the hydrogen uninvestigate the effects of compartmentalized geometries indicating a maximum-to-average concentration ratio of less than 1.2. One arrives at similar conclusions even when one considers the concentration gradient in the region above the 23 m source elevation.

The impact of a simulated large LOCA with gas injection at the 13 m elevation was studied in Test T31.5. This experiment indicated the potential for an initially stratified mixture to develop in HDR with higher hydrogen concentrations in the dome. However, the difference in hydrogen concentration was small. After about 3 hours the dome and source elevations had hydrogen concentrations within about 40 % of one another (maximum to containment average of about 1.2). After 10 hours the initially stratified mixture reached near uniformity.

B. Effect of Spray on Hydrogen Concentration

Test E11.2 indicated the effect of spraying into a stratified mixture can cause a significant increase in the local steam concentration in the gas rich region at the expense of a depletion of the gas in the low concentration region. In essence, spraying could in that limited situation make a bad situation worse. This behavior was a result of a quiescent steam condensation driven flow to the upper compartment which due to the poor natural convective patterns in the HDR and initial stratified behavior could not readily remix with the rest of the containment. Similar testing performed for an initially well mixed situation is designated test E11.4. For this test the condensation process resulted in a uniform gradual rise in the gas concentration throughout the containment.

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NUPEC Hydrogen Distribution Test¹⁸

A hydrogen distribution test facility has been developed by NUPEC (Nuclear Power Engineering Corporation) to study hydrogen distribution in prototypical containments. To accomplish this goal NUPEC designed a 1/4 linearly scaled steel containment with 25 compartments (each compartment representing a room in the actual containment). The NUPEC containment vessel has a volume of 1600 m³, a diameter of 10 m and a height of 20 m. Hydrogen gas was simulated by helium. Tests included steam injection as well. Thirty five experiments were conducted through March of 1992. Helium concentration was measured in every compartment and in several places within the upper dome. Details of these tests are currently unavailable. However, based on preliminary information and summary reports from the experimenters the following conclusions were drawn.

1. Several mixing loops were formed by natural convection. These flows were sufficient to prevent local "gas" concentration hot spots provided the source of injection was the lower compartment.
2. Helium gas sampling in the dome (whose volume is 70 % of the total facility volume) showed almost complete uniformity.
3. Containment spray operation enhances natural convection processes.

4.3 EXPERIMENTAL EVIDENCE SUPPORTING CONTROL OF HYDROGEN VIA DELIBERATE IGNITION SYSTEMS

Considerable experimentation has been performed with the intent of investigating the efficacy of using various forms of igniters in controlling the hydrogen concentration in hydrogen-air-steam mixtures. The overall goal of these tests were to validate the hydrogen igniter system design selected by the ice condenser PWRs and General Electric Mark III BWRs. These tests were conducted at varying scales with various degrees of simulation. Most tests focused on the GMAC-7G glow plugs. Table 4.3-1 summarizes the most significant of the igniter tests performed in the United States. Additional supportive tests were also performed in Europe and Japan.

The test program provided valuable insights into the mechanisms associated with hydrogen burning including information associated with hydrogen placement and combustion efficiency. Based on a review of these experiments it was concluded that igniters can limit hydrogen concentrations to the 4 to 7 volume percent condition. Furthermore, several general rules were developed for igniter placement which were adopted in the System 80+ Hydrogen Mitigation System design (See Section 5).

SNL FITS TESTS¹⁹

These tests consisted of a series of 239 hydrogen:air:steam experiments performed at Sandia National Laboratory's (SNL's) 5.6 m³ Fully Instrumented Test Site (FITS). These experiments addressed the

flammability limits of combustible atmospheres that might occur inside containment during severe accidents. Tests investigated mixture responses for hydrogen:steam: air mixtures with volume concentrations up to 70 volume percent hydrogen and about 56 volume percent steam.

These tests were highly successful in defining the hydrogen:air:steam flammability triangle. In addition to the limit criteria, tests also investigated combustion completeness issues, effects of mixture temperature and the resultant burn pressure rise. Pertinent conclusions from this program were as follows:

- 1) Hydrogen:air: steam mixtures will be inert to combustion provided the steam concentration exceeds 52%.
- 2) Pressures predicted using assumptions Adiabatic Isochoric Complete Combustion (AICC) bound experimental predictions.
- 3) Combustion of high volume percent hydrogen mixtures (up to 30 volume per-cent) was consistently observed to result in a deflagration. In fact of 70 experiments performed at hydrogen concentrations greater than 13 volume percent, no detonations were observed.

SNL VGES TESTS¹⁹

The Variable Geometry Experimental System (VGES) combustion chamber is used extensively at SNL for studies of closed volume deflagrations of hydrogen:air mixtures. The purpose of these experiments were to establish igniter performance and determine the effect of diluents and water sprays on hydrogen deflagration. Three igniter designs were studied, these included: exposed 300 W photolamp filament, a 30-J raised spark gap and the GMAC 7G standard glow plug and the TAYCO Model 193-3442-4 helical igniter. This later device is currently used for existing deliberate ignition systems.

Approximately 100 tests were performed in all. Of primary importance to these tests include observations with regard to igniter performance. These findings are as follows:

- 1) Peak combustion pressures were observed to rise rapidly as the hydrogen concentration reached 5 to 8%. Predictions of these burns via AICC methods, suggest the combustion pressures are bounded by AICC.
- 2) Igniter performance was found to have a significant influence on all hydrogen concentrations below 8 volume percent.
- 3) Ideal gas inerting via nitrogen and carbon dioxide were considered in various mixtures. Based on these studies the diluent mixture for CO₂ sufficient to inert the mixture was 54%.
- 4) Tests of igniter performance in the presence of a water spray and high velocity draughts, suggest that direct spray impingement upon 53 l/m² min can defeat the glow plug.

NTS TESTS¹⁹

A series of tests to investigate hydrogen combustion in a large scale facility were conducted by EG&G at the Nevada Test Site (NTS). These tests investigated hydrogen mixing and survivability of safety related equipment in postulated degraded core accident hydrogen burn environments. Tests included burns in premixed mixtures and continuous injection hydrogen sources. Approximately 40 experiments were performed. For premixed tests, the mixture composition was varied over a range of 5 to 13 % hydrogen and 4 to 40% steam. Ignition for these test was initiated by a GMAC 7G glow plug, located at various positions within the vessel. The NTS vessel was 15.85 m in diameter, with a volume of 2048 m³.

Test results provided considerable practical information on igniter performance and placement. Tests clearly indicated that combustion completeness is directly correlated with igniter placement. Location of igniters towards the top of the vessel limited upward burning and shifted the flammability limit to that of downward burning which requires higher hydrogen concentrations. It was also noted that the glow plug could effectively ignite hydrogen mixtures as low as 5.2 volume per cent (provided steam concentrations are low). Ignition above 8 volume per cent hydrogen consistently resulted in complete combustion.

LLNL HYDROGEN IGNITER EXPERIMENTAL PROGRAM¹⁹

This program consisted of an NRC directed effort to investigate the effectiveness of glow plugs as effective deliberate ignition sources in hydrogen:steam:air mixtures. Approximately 100 experiments were performed. Tests were conducted at a pressure of 1.4 MPa in a vessel with a 0.3 m³ free volume. The facility investigated the GM AC 7G glow plug which was positioned at various locations within the test vessel.

These experiments provided additional confirmation that AICC prediction methods would effectively bound hydrogen burn pressures for hydrogen burns up to 16 volume per cent hydrogen. Furthermore, hydrogen burns could be achieved at concentrations as low as 6 volume percent and that complete combustion was expected at hydrogen concentrations of 8 volume percent. These igniter tests also noted that the GMAC 7G glow plug consistently ignited mixtures at surface temperatures between 700 and 800 °C, with higher temperatures necessary to ignite steam mixtures, and "showed no appreciable deterioration throughout the series of tests".

Experiments also confirmed that steam concentrations in excess of 50 volume percent can effectively inert the combustion process. Tests performed in this test series investigated the combustion characteristics of an initially steam inerted mixture as the mixture condensed. These tests were performed with a 10% hydrogen concentration while the glow plug was activated. The condensation process was noted to result in the consumption of hydrogen without a consequent discrete pressure rise.

WHITESHELL TESTS^{19,20}

The Electric Power Research Institute (EPRI) sponsored a series of over 300 premixed hydrogen combustion tests at the Whiteshell Nuclear Research Establishment in Pinawa, Manitoba. The program was focused in confirming the effectiveness of a deliberate ignition system in controlling hydrogen during a severe accident.

Tests were conducted in a 17 liter quasi-spherical facility. Two types of glow plugs were studied: the GMAC 7G glow plug and the Tayco Model 193-3442-4 helical igniter.

These tests showed that either of the igniters could effectively ignite dry hydrogen:air mixtures in quiescent conditions at 5.5 volume percent hydrogen, down to 4.5 volume percent under turbulent conditions. Furthermore, mixtures with as much as 55 volume percent steam were ignited in both quiescent and turbulent tests.

Assessment of combustion completeness indicate that for hydrogen mixtures greater than 9 volume percent burns were essentially complete. Combustion completeness was noted to also be dependent upon the steam concentration. Significant reduction in combustion completeness was observed for steam concentrations greater than 30 %.

The Whiteshell experiments also investigated combustion in an initially inerted steam environment subject to steam condensation. In this experiment an initially inerted steam:hydrogen:air mixture was cooled to bring the mixture into the flammable range. Ignition was typically observed to occur once the mixture passed through the deflagration ignition limit (See Figure 4.3-1). These ignitions occurred well in advance of any potential detonation.

ACUREX TESTS¹⁹

The Acurex corporation conducted a series of intermediate scale hydrogen combustion experiments. The program was conducted in a 17.83 m³ vessel. Both premixed and continuous hydrogen injection tests were performed. The specific objectives of these tests were to investigate effects of injection rates (hydrogen and steam), igniter location and water sprays and fogs on deliberate ignition.

These tests concluded that location of the igniter can affect the effectiveness of the deliberate ignition system. Injection of hydrogen above the igniter location will result in that igniter being bypassed. Igniters located near the top wall of the vessel would ignite, however, only after the hydrogen concentration reached the hydrogen concentration sufficient for downward flame propagation. The presence of sprays tended to produce longer burns with a smaller pressure rise.

FENWAL TEST¹⁹

The Fenwal tests were performed for Westinghouse Electric Corporation and several utilities to determine the effectiveness of glow plug igniters in

a deliberate ignition system. The test facility was a small scale 3.8 m³ spherical vessel. Tests included dry and wet hydrogen:air mixtures with both premixed and continuous injection hydrogen sources. Hydrogen concentrations ranged from 5 to 12 % by volume with steam concentration up to 40 volume percent. All tests were conducted with a GMAC 7G glow plug.

At low hydrogen concentrations, the effect of sprays and fans were to increase the hydrogen combustion pressure rise. This is presumably a result of increased mixing within the facility. At hydrogen concentrations about 8 volume percent, the hydrogen burn was essentially complete.

The tests indicated that upward burns would propagate at hydrogen concentrations as low as 4 volume percent; at 6.5 % the burn will propagate sideways and at 8.5 % the burn propagates in all directions.

NUPEC HYDROGEN IGNITER TESTS^{21,33}

NUPEC has embarked on Containment Integrity Project for proving the reliability of Reactor Containment Vessels. This project has been ongoing since June 1987. This program includes both large (270 m³) and small (5 m³) combustion experiments. Small scale tests were performed to investigate hydrogen combustion phenomena and flame transition phenomena. Results from these tests were generally consistent with early data obtained from similar programs in the United States. Large scale test data will simulate multicompartment features of an actual plant.

UNIVERSITY OF PISA HYDROGEN IGNITER TESTS²²

This test series was conducted by the Department of Mechanical and Nuclear Constructions at the University of Pisa and the ENEA (Comitato Nazionale per la Ricerca e per lo Sviluppo dell' Energia Nucleare e delle Energie Alternative). These experiments were directed towards establishing deflagration characteristics and examining the capabilities of igniters for hydrogen control. Tests were conducted in the 0.5 m³ Hydro-SC facility at the University of Pisa. Tests employed glow plug igniters and included hydrogen concentrations from 4 to 16 volume percent with and without spray injection.

The Hydro-SC tests confirmed results of similar experiments performed in the United States. The test further indicated that the igniter performed its function in the presence of water sprays.

MARK III DEMONSTRATION TEST^{19,23,25,26,27}

The Hydrogen Control Owners Group (HCOG) sponsored a 1/4 -Scale Mark III Containment Combustion Hydrogen Program to determine the thermal environment to which critical plant equipment in the Mark III containment may be subjected to as a result of hydrogen combustion following a severe accident. The tests were conducted by the Factory Mutual Research Corporation. The facility was designed using Froude Number scaling techniques to simulate the details of the containment systems having an

important impact on the modeling of the combustion phenomenon. These features included a simulation of the HCOG deliberate ignition system, a model of the SRV sparger geometry and suppression pool, containment sprays and fan coolers.

The test simulated three types of transient hydrogen releases. Hydrogen could be released via a simulated automatic depressurization system (ADS), a stuck open relief valve into a sparger, or a LOCA event directly into the containment.

The experimental program, in essence, was a scaled demonstration of the efficacy and practicality of a distributed deliberate ignition system for use in degraded core and severe accident hydrogen control. Prior to the onset of this program the experimenters believed that the igniter system would primarily control hydrogen accumulation via a series of discrete deflagrations. It was also recognized that steady diffusion flames may develop for certain scenarios. Instead, the test program indicated that the dominant mode of combustion was the diffusion flame. Unburnt hydrogen was controlled to levels equivalent to 4.5 to 5 % by volume on a dry basis. Diffusion flames were observed to be anchored to the surface of the suppression pool. No significant pressure excursions were noted. Consequently it was concluded that "the occurrence of successive deflagrations do not appear possible as a major mechanism of hydrogen consumption if a distributed ignition system is activated in the containment volume".

Tests were performed using two hydrogen release histories typical of core degradation associated with a severe accident. These profiles were scaled to represent an initially rapid hydrogen release over 1600 seconds with a hydrogen release rate peak in the range of .24 to .48 kg/sec. A smaller tail hydrogen release rate was also modeled with prototypic constant production rate of .07 kg/sec for 11,200 seconds. These values are generally typical of a realistic core degradation process in System 80+. In the System 80+ design the zircaloy content of the core is about 35,000 kg. Hydrogen will be rapidly released during a core degradation process over a period of about one half hour. In this time between 30 and 50 percent of the core can be oxidized. Thus, hydrogen release rates will be on the order of .26 to .43 kg/sec. These hydrogen release rates typical of the FMC test are in close agreement for a similar driving function for System 80+.

Tests with Hydrogen Released Through the Suppression Pool

Several experiments were performed with hydrogen released through the suppression pool. In these circumstances the original ignition was observed at the HCU floor (ceiling above the suppression pool). The flame ultimately anchored to the suppression pool surface as a steady diffusion flame. As oxygen depleted from these regions the flame was observed to lift to the HCU floor. Several aspects of these test observations are consistent with System 80+. First the IRWST in System 80+ is expected to ultimately be an oxygen depleted region. Consequently diffusion flames on the pool surface per se are not expected. However, anchoring of diffusion flames at the exit of the IRWST vents is expected. Should the flame lift from the IRWST vents, the flame will expand into the SG chimney areas and be transported to the upper containment as a hydrogen depleted

mixture. Diffusion flames were observed to maintain the containment concentration to about 5 volume percent. Steam concentrations during these tests were about 10-15 volume percent.

Impact of Reduction in Igniter Availability ²⁶

HCOG test S-13 investigated the importance of the number of igniters on the overall system performance. In this experiment only 18 of the 48 simulated system igniters were powered. Based on a comparison of S-13 to its counterpart experiment the HCOG investigators concluded that "deactivation of 29 ... igniters had no significant effect on combustion phenomena".

Observations Regarding Mixing ²⁶

HCOG experiments resulted in "excellent" hydrogen mixing both before and after hydrogen ignition. A review of the data showed no evidence of localized hydrogen accumulations.

TABLE 4.3-1 : SUMMARY OF PERTINENT IGNITER TEST INFORMATION				
TEST SERIES	MIXTURE	VOLUME (M ³)	GLOW PLUG TESTED	EXPERIMENTAL FINDINGS
SNL FITS	H ₂ /AIR/ STM	5.6	Y	<p>FULLY INSTRUMENTED TEST FACILITY (FITS)</p> <p>TEST OBJECTIVES:</p> <p>ASSESS COMBUSTION CHARACTERISTICS AND FLAMMABILITY LIMITS OF SEVERE ACCIDENT CONTAINMENT ATMOSPHERES</p> <p>RESULTS:</p> <ol style="list-style-type: none"> 1) DEFLAGRATIONS OBSERVED FOR HYDROGEN:AIR MIXTURES WITH UP TO A GREATER THAN 30 VOLUME PERCENT HYDROGEN CONCENTRATION. NOTE: MORE THAN 70 HYDROGEN:AIR BURNS WERE OBSERVED AT CONCENTRATIONS GREATER THAN 13 V/O WITHOUT THE INITIATION OF A DETONATION. 2) DETAILED FLAMMABILITY CURVE DEVELOPED FOR H₂:AIR:STEAM MIXTURES. TESTS INDICATED THAT STEAM CONCENTRATIONS GREATER THAN 52 V/O INSERTED BURNING 3) AICC PREDICTED PRESSURES BOUNDED OBSERVED PRESSURIZATION
VGES	H ₂ /AIR	5.1	Y	<p>VARIABLE GEOMETRY EXPERIMENTAL SYSTEM (VGES)</p> <p>TEST OBJECTIVE:</p> <p>PARAMETRICALLY STUDY CLOSED VOLUME DEFLAGRATIONS IN HYDROGEN: AIR MIXTURES</p> <p>RESULTS:</p> <ol style="list-style-type: none"> 1) PREDICTED AICC PRESSURES BOUNDED ALL OBSERVED HYDROGEN BURNS (HYDROGEN CONCENTRATION TESTED UP TO 24 V/O) 2. IGNITER LOCATION IS IMPORTANT FOR THE IGNITION OF HYDROGEN:AIR MIXTURES BELOW 8 V/O. AT THESE LOWER CONCENTRATIONS MOVEMENT OF IGNITER UPWARD REDUCED BURN COMPLETENESS IN THE VOLUME SINCE FLAME PROPAGATED UPWARD. 3. A GM-AC GLOW PLUG NEEDED A SURFACE TEMPERATURE OF ABOUT 1330 °F TO IGNITE A LEAN HYDROGEN:AIR MIXTURE 4. WATER SPRAYS GREATER THAN ABOUT 40 L/M2/MIN CAN RENDER AN UNSHIELDED GLOW PLUG INEFFECTIVE. GLOW PLUG PERFORMANCE CAN BE ASSURED BY SHIELDING THE PLUG FROM DIRECT WATER IMPINGEMENT.

TABLE 4.3-1 : SUMMARY OF PERTINENT IGNITER TEST INFORMATION				
TEST SERIES	MIXTURE	VOLUME (M ³)	GLOW PLUG TESTED	EXPERIMENTAL FINDINGS
NTS	H ₂ /air /stm	2048	Y	<p>NEVADA TEST SITE (NTS)</p> <p>TEST OBJECTIVE:</p> <p>STUDY HYDROGEN MIXING AND IGNITION PROCESSES AND THE SURVIVABILITY OF SAFETY RELATED EQUIPMENT IN A LARGE SCALE FACILITY ASSOCIATED WITH BURNS IN A HYDROGEN:AIR:STEAM MIXTURE</p> <p>RESULTS:</p> <ol style="list-style-type: none"> 1) COMBUSTION COMPLETENESS IS ASSOCIATED WITH IGNITER PLACEMENT AT HYDROGEN CONCENTRATIONS BELOW 7 V/O 2) GLOW PLUGS COULD EFFECTIVELY IGNITE MIXTURES DOWN TO 5.2 HYDROGEN VOLUME PERCENT 3) IGNITION ABOVE 8 V/O WAS ASSOCIATED WITH COMPLETE COMBUSTION
LLNL	H ₂ /air /stm	0.3	Y	<p>LLNL EXPERIMENTAL PROGRAM</p> <p>TEST OBJECTIVE:</p> <p>EVALUATE USE OF THE GM AC 7G GLOW PLUG AS A DELIBERATE IGNITION SOURCE FOR HYDROGEN:AIR:STEAM MIXTURES</p> <p>STUDY COMBUSTION PHENOMENOLOGY AND EFFECTS OF STEAM AND WATER FOGS.</p> <p>RESULTS:</p> <ol style="list-style-type: none"> 1) AICC PREDICTIONS BOUNDED PRESSURES FOR ALL BURNS INCLUDING HYDROGEN CONCENTRATIONS UP TO 16 V/O IN DRY AIR. 2) HYDROGEN BURNS WERE ACHIEVED AT CONCENTRATIONS ABOVE 6 V/O. COMPLETE BURNS OCCURRED AT 8 V/O. 3) VOLUMETRIC STEAM CONCENTRATIONS IN EXCESS OF 50% CAN INERT THE BURNING PROCESS 4) CONDENSATION TESTS WITH 10 V/O HYDROGEN DID NOT RESULT IN A NOTICEABLE BURN EVEN THOUGH THE IGNITER CONSUMED THE HYDROGEN AS CONDENSATION PROCEEDED BELOW THE 50% THRESHOLD. 5) GM AC 7G GLOW PLUG CONSISTENTLY IGNITED MIXTURES AT SURFACE TEMPERATURES BETWEEN 700 AND 800 C AND SHOWED NO SIGNIFICANT DETERIORATION

TABLE 4.3-1 : SUMMARY OF PERTINENT IGNITER TEST INFORMATION

TEST SERIES	MIXTURE	VOLUME (M ³)	GLOW PLUG TEST-ED	EXPERIMENTAL FINDINGS
white-shell	H ₂ /air/stm	17 liter	Y	<p>TEST OBJECTIVE</p> <p>CONFIRM EFFECTIVENESS OF A DELIBERATE IGNITION SYSTEM IN CONTROLLING HYDROGEN RELEASED DURING A SEVERE ACCIDENT</p> <p>RESULTS:</p> <ol style="list-style-type: none"> 1) FOR DRY HYDROGEN:AIR WITH MORE THAN 5.5 V/O UNDER QUIESCENT CONDITIONS, DRY CONCENTRATIONS AS LOW AS 4.5 V/O COULD BE IGNITED UNDER TURBULENT CONDITIONS. 2) GLOW PLUG IS CAPABLE OF IGNITING MIXTURES AS MUCH AS 78 V/O HYDROGEN 3) PRESENCE OF STEAM REDUCES COMBUSTION COMPLETES PARTICULARLY 4) STEAM CONDENSATION EXPERIMENTS INDICATED THAT DEFLAGRATIONS WILL OCCUR PRIOR TO REACHING A DETONABLE CONDITION
ACUREX		17.8		<p>TEST OBJECTIVE</p> <p>INVESTIGATE EFFECTS OF HYDROGEN AND STEAM FLOWRATES, IGNITER LOCATION AND WATER SPRAYS ON THE DELIBERATE IGNITION OF FLAMMABLE CONTAINMENT ATMOSPHERES.</p> <p>RESULTS</p> <ol style="list-style-type: none"> 1) LOCATION OF IGNITER EFFECTED COMPLETENESS AND TIMING OF COMBUSTION. BOTH LOCATION OF IGNITERS AT THE BOTTOM AND TOP OF THE FACILITY PROVIDED DISCRETE BURNS WITH GREATER MAGNITUDE. LOCATION OF IGNITERS AT THE MID-SPAN RESULTED IN CONTINUOUS LOW PRESSURE BURNING OF HYDROGEN.
FENWAL				<p>TEST OBJECTIVE</p> <p>TEST OF THE PERFORMANCE AND DURABILITY OF THE GM AC 7G GLOW PLUG-SHIELD SYSTEM TO ACT AS A DELIBERATE IGNITION SYSTEM. HYDROGEN CONDITIONS RANGED FROM 5 TO 12 VOLUME PERCENT.</p> <p>RESULTS</p> <ol style="list-style-type: none"> 1) AT LOW CONCENTRATIONS (4%) BURNS PROPAGATE DOWNWARD AT 6% BURNS PROPAGATE SIDEWAYS AT 8.5% BURNS PROCEED IN ALL DIRECTIONS

TABLE 4.3-1 : SUMMARY OF PERTINENT IGNITER TEST INFORMATION

TEST SERIES	MIXTURE	VOLUME (M ³)	GLOW PLUG TEST-ED	EXPERIMENTAL FINDINGS
Mark III				<p>TEST OBJECTIVE</p> <p>PROVIDE A SCALED DEMONSTRATION FOR THE APPLICABILITY OF A DELIBERATE IGNITION SYSTEM TO THE MARK III BWR CONTAINMENT.</p> <p>RESULTS</p> <p>1) THE IGNITER SYSTEM WAS CAPABLE OF MAINTAINING THE HYDROGEN CONCENTRATION IN THE CONTAINMENT TO 4-5% (ON A DRY BASIS) EVEN WHEN THE OXYGEN LEVEL REDUCED TO 7.2 VOLUME PER-CENT, REGARDLESS OF THE MAGNITUDE OF THE HYDROGEN RELEASE.</p> <p>2) IGNITION PRESSURE TRANSIENTS WERE MODEST, WITH NO PRESSURE EXCURSIONS AND IGNITION USUALLY OCCURRED AT CONTAINMENT AVERAGE HYDROGEN LEVELS OF 1-2 %</p>

*GM-AC 7G GLOW PLUG

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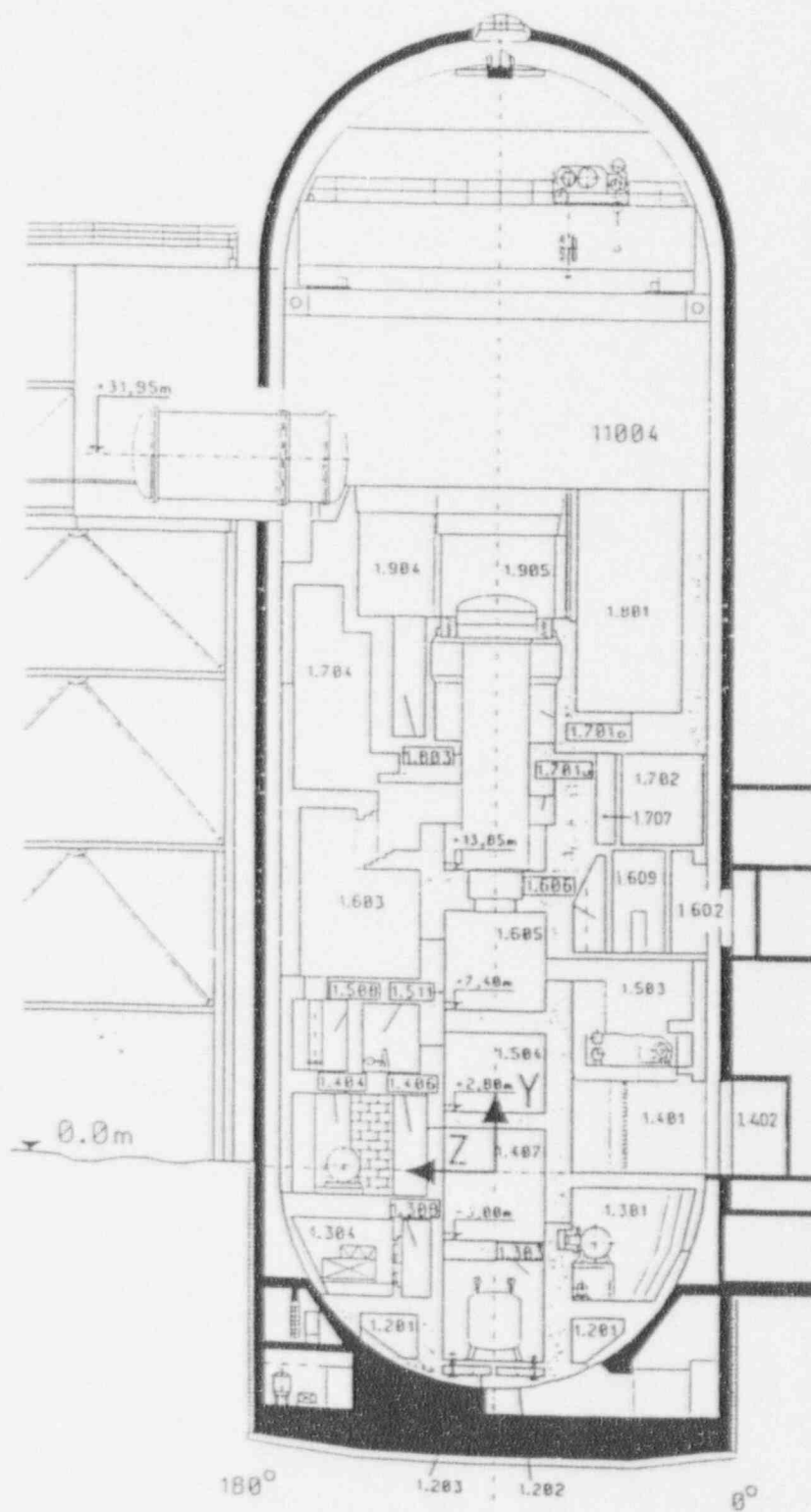


FIGURE 4.2-1: HDR FACILITY

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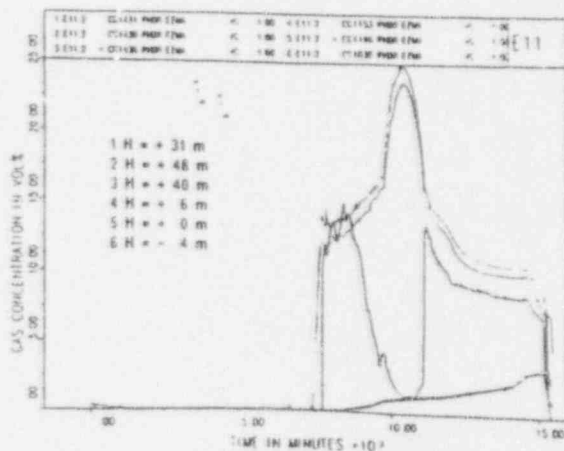


FIGURE 4.2-2: HDR TEST E11.2 RESULTS
GAS CONCENTRATION ALONG STAIRCASE
Test shows concentration differential between lower and upper HDR regions (above and below point of injection)

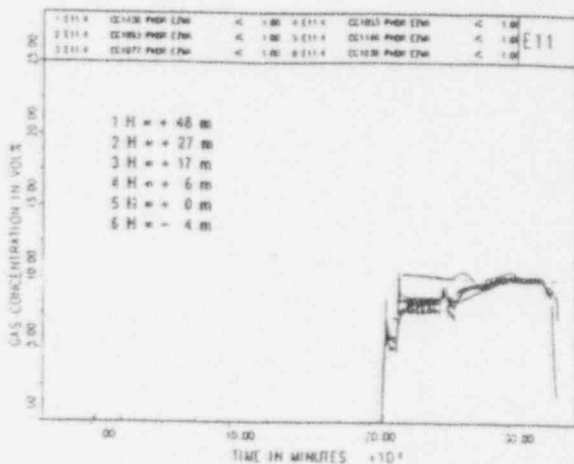


FIGURE 4.2-3: HDR TEST E11.4 RESULTS
GAS CONCENTRATION ALONG STAIRCASE
Test shows good mixing throughout the HDR facility

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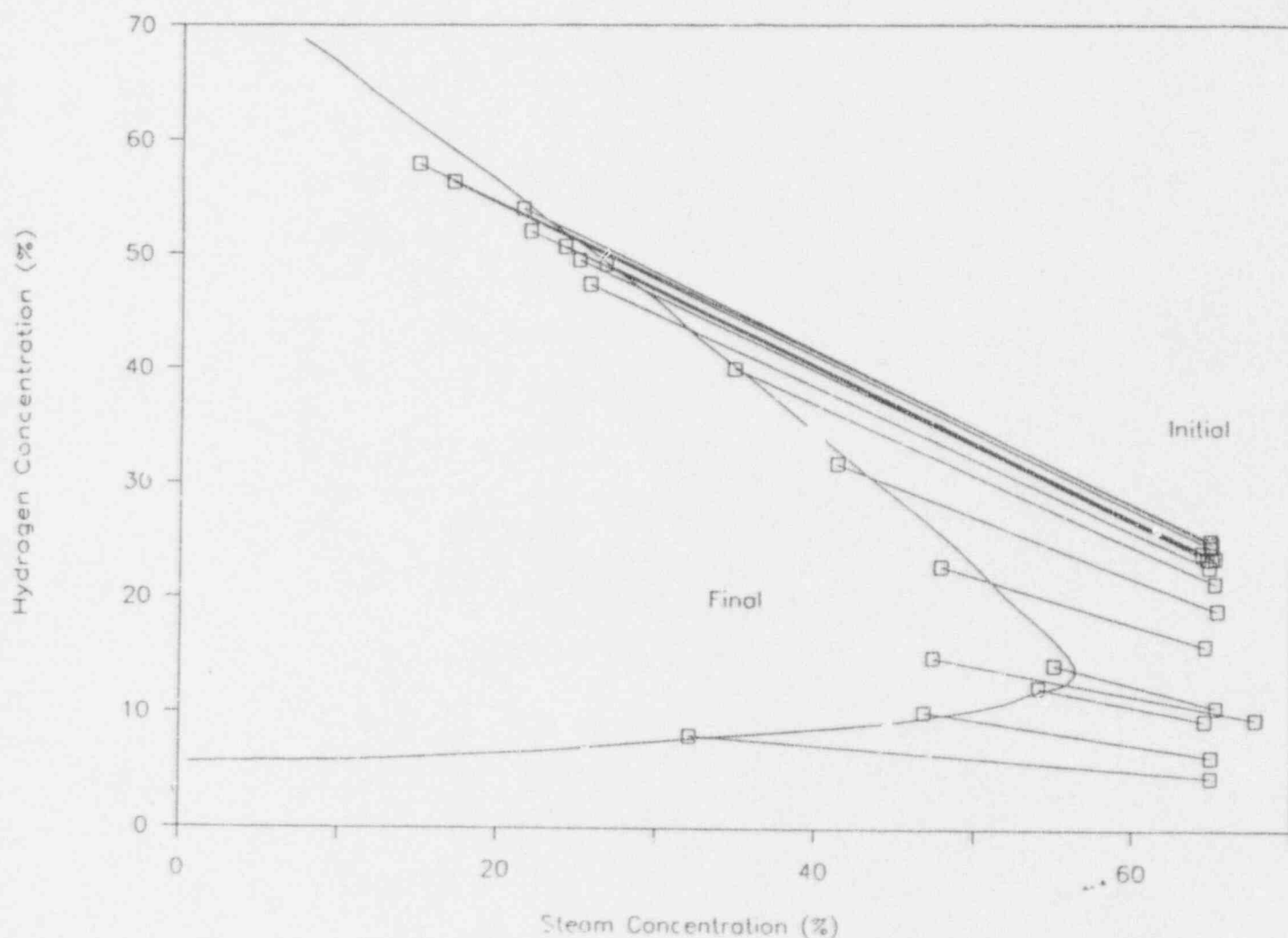


FIGURE 4.3-1: Hydrogen Ignition limits for steam condensing atmospheres (Results of Whiteshell Steam Condensation Test)

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5.0 HYDROGEN IGNITER PLACEMENT GUIDELINES FOR SYSTEM 80+

This section summarizes the goals and guidance used in establishing the design for the System 80+ Hydrogen Mitigation System (HMS). The HMS consists of a system of igniters installed in the containment to promote the combustion of hydrogen in a controlled manner so as to maintain the average containment hydrogen concentration below the threshold value for potential detonation.

5.1 HMS DESIGN GOALS

The cost effectiveness of HMS for existing large dry PWRs (Zion and Surry) have been studied by the NRC. Based on these studies the hydrogen igniters were not found to be a cost effective accident mitigation measure for either plant. This conclusion was based on several factors including low zircaloy content in the core (limited hydrogen production potential), and relatively low core damage frequency for these plants and requirements associated with the "backfit rule". Even stronger arguments regarding the "low cost-benefit" of the HMS can be advanced for System 80+. System 80+ has a core melt frequency (on the order of 1×10^{-6} per reactor year) and a low conditional containment failure probability (less than 0.10). However, since System 80+ is an evolutionary ALWR the HMS is included in the plant design basis.

In the design of the HMS a number of high level design goals were established. These goals were established so that the scope of the HMS could be clearly defined. The HMS design goals are:

1. The igniter system is designed such that the global hydrogen concentration will be below 8 v/o and local hydrogen concentrations for containment volumes away from the hydrogen source can be maintained below 10 v/o.

In the event no detonations occur, containment integrity is assured since the peak AICC burn pressures is below the ASME service Level C limit for the System 80+ containment (See CESSAR-DC Section 19.11.4).

2. In the event local concentrations in containment sub-volumes or small rooms exceed 10 v/o near the hydrogen source, the resulting mixture is either
 - (a) not detonable (either via steam inerting or oxygen depletion) or,
 - (b) a detonation in the region will not result in a threat to containment integrity.

The first criteria limits the global threat to the containment to well within the structural capability. The second goal is intended to ensure that containment integrity is not compromised on a local basis.

Based on limited testing of the Mark III igniter system, the above goals appear to be easily achievable, and in fact, the actual HMS is expected to limit hydrogen concentrations in the average range of 5 to 7 volume percent.

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5.2 HMS PLACEMENT CRITERIA

In order to design the HMS several issues were addressed to ensure that the system is (1) reliable (2) maintainable and (c) cost-effective. Additional placement criteria were established based on observations of igniter and hydrogen distribution experiments described in Section 4 above. The HMS design criteria are summarized in Table 5.2-1 and are briefly discussed below.

Reliability

The issue of reliability includes system availability issues such as alternate power supplies, and redundancy. Details of the Igniter System can be found in Section 6.2.5 of the CESSAR-DC. To ensure the HMS is highly reliable, the following criteria were established:

1. Igniters should be available in redundant pairs.

Igniters are to be placed, in general, in regions with at least one pair of igniters in each designated region where hydrogen control is desired. Paired igniters will be powered from separate power supply divisions. The intent of the redundant system is to control containment hydrogen concentration with only one-half of the HMS igniters operational.

2. HMS power will be diverse and redundant

This is accomplished by providing power to the HMS igniters via offsite power, emergency diesels and the combustion turbine generator. A minimal subset of the HMS igniters (approximately 32 igniters) is to be powered off Class 1E Division batteries. Sufficient power is available in the station batteries to ensure operation of one-half of the HMS igniters for a period of four hours. As identified in Item 1, partial operation of the HMS will be adequate to accomplish the System 80+ hydrogen control objective of preventing a potential detonation within the containment following a severe accident.

Maintainability

Experience on the Duke Ice Con and PWR units have demonstrated that cost-efficiency of the HMS will be associated with ability of the plant staff to maintain and test the igniters. The System 80+ igniters have been located with consideration of maintainability. Maintainability criteria are associated with:

1. Locating igniters in accessible (and low radiation) areas and on existing walls
2. Assuring that all igniters can be tested and replaced without excessive manpower costs
3. limiting igniter placement in the IRWST

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Technical Placement Criteria

The above criteria provide overall guidance in the general manner in which the HIS igniters are to be placed. In this section the specific criteria for placement of the igniters within the containment are defined along with a brief discussion of the basis.

1. Flowpath Requirements

The System 80+ igniter placement guidelines are defined in Table 5.2-1. Several criteria specifically relate to the placement of the igniter with respect to the system flowpaths and sources. These criteria require the placement of igniters:

1. along dominant flowpaths
2. above hydrogen sources
3. along secondary flowpaths
4. at multiple burn levels

These criteria are generally derived from a combination of engineering judgement supported by experimental evidence and plant analyses. Hydrogen ignition tests discussed above have generally indicated that to be effective igniters should be above the hydrogen source. This will maximize the hydrogen consumption while maintaining the global hydrogen concentration in the containment low. This was observed in the Mark III 1:4 scale experiments as well as the large scale NTS facility.

In addition the above criteria also suggest the importance of identifying and providing ignition sources along important flow paths. The primary need for igniting these regions is to aid in the consumption of hydrogen which, either due to the initial igniter location or local steam inerting effects, is not initially burned. This hydrogen will be transported to upper containment regions. In the System 80+ design, the dominant flowpath for hydrogen transport will be up through the "chimney" created by the steam generator enclosures. Secondary flowpaths may also develop which connect the reactor cavity to the upper containment through the reactor cavity annulus and manway. Thus, ignition sources are to be provided along these paths.

2. consideration of enclosures

Enclosures may become regions of high hydrogen concentration. Typically enclosed regions are a concern if they can become the source of a hydrogen release. All enclosures in the System 80+ containment are vented. In order to ensure detonable hydrogen accumulations do not develop all System 80+ enclosures will be supplied with a pair of igniters.

3. igniter spacing and location below ceilings

General rules for igniter spacing and placement were established based on a review of existing hydrogen ignition data. These rules were:

- a. igniters can be separated by 50-75 feet

This allows a single igniter pair placement per floor in the steam generator compartment and minimizes the number of igniters required in the upper dome. NTS tests demonstrate that the hydrogen concentration in large open regions can be controlled by a single

igniter. The typical volume in the NTS test facility was about 75,000 ft³ with a diameter of about 50 feet.

b. igniters should be located 10 feet below the ceiling

This criteria encourages upward burning and maximizes the per igniter hydrogen consumption in the vicinity of the source. This recommendation is based on observations in various hydrogen ignition tests which noted upward burning occurred at lower hydrogen concentrations than either sideways or downwards burning. Consequently, when feasible igniters should be placed many feet below obstructions (such as ceilings). Typically, 10 feet is expected to allow sufficient to allow upward combustion to occur.

TABLE 5.2-1: SUMMARY OF SPECIFIC IGNITER PLACEMENT AND DESIGN CRITERIA FOR SYSTEM 80+				PRIOR ITY
NO.	CRITERIA	BASIS	NOTES	
1	IGNITERS PLACED ALONG DOMINANT NATURAL CIRCULATION FLOW PATTERN PATHWAY.	SYSTEM 80+ IS DESIGNED WITH A SINGLE DOMINANT CHIMNEY FLOWPATH WITH THE SG ENCLOSURES AND INTERIOR CRANE WALL SERVING AS THE LOW DENSITY RISER AND THE OUTER CRANE ANNULUS SERVING AS THE DENSER FLUID RETURN DOWNCOMER. THE NATURAL CIRCULATION PATTERN IS DRIVEN BY VERY LOW PRESSURE DIFFERENTIALS. SINCE ALMOST ALL THE HYDROGEN WILL FLOW THROUGH THIS PATHWAY DOMINANT REGIONS IN THE UPFLOW PORTION OF THE CHIMNEY ARE REQUIRED TO HAVE IGNITERS. DOMINANT FLOWPATH CONFIRMED VIA DETAILED THERMAL HYDRAULIC ANALYSIS.	FIGURE 5-1 A, I	2
2	IGNITERS ARE PLACED IN VICINITY AND ABOVE HYDROGEN SOURCES.	TYPICAL EXPECTED SOURCES OF HYDROGEN INCLUDE ALL RCS PRIMARY PIPING, NON-ISOLABLE CONNECTING PIPING AND IRWST VENTS	A, B	1
3	IGNITERS LOCATED IN CLOSED AND LESS WELL VENTED REGIONS.	DEAD REGIONS ALLOW POTENTIAL FOR HYDROGEN TO ACCUMULATE.	C	7
4	MULTIPLE LEVELS OF BURNING IN DOMINANT FLOW PATHS.	MULTILEVELED BURNING WILL MINIMIZE THE RISK OF LOCALIZED STEAM INERTED REGIONS FROM PREVENTING HYDROGEN COMBUSTION AT THE IGNITERS. MULTILEVELEDURNS ALSO ALLOW BURNING OFF OF ADDITIONAL HYDROGEN THAT WAS NOT PREVIOUSLY BURNED DUE TO INCOMPLETE COMBUSTION AT LOWER LEVELS. THIS METHOD WILL ALSO RESULT IN HIGHER TEMPERATURES IN THE DOMINANT UPFLOW PATHS AND INCREASE CIRCULATION THROUGH THE MULTILEVELED FLOW PATHS.	G	3
5.	AXIAL SPACING OF MULTI-LEVEL IGNITERS BASED ON FLOOR SPACING.	NTS DATA SUGGESTS THAT IGNITERS CAN CONTROL HYDROGEN CONCENTRATION IN VOLUMES WITH VERTICAL HEIGHTS GREATER THAN 50 FEET. THIS DISTANCE IS TYPICALLY LARGER THAN THE SYSTEM 80+ FLOOR SEPARATION.		4
6.	HIGHLY RELIABLE POWER SOURCE FOR MINIMUM IGNITER SET.	OPERATING EXPERIENCE SUGGESTS THAT IGNITER FAILURES MAY OCCUR DURING PLANT OPERATION. THEREFORE, POWER TO BOTH THE MINIMUM AND SUPPLEMENTAL HYDROGEN SET SHOULD BE HIGHLY RELIABLE TO PROVIDE REASONABLE ASSURANCE THAT PERFORMANCE GOALS ARE ACHIEVED.	F, J	16
7.	IGNITER LOCATIONS SUPPORTED BY AN IGNITER PAIR IN THE SAME GENERAL VICINITY.	THIS CRITERIA PROVIDES REDUNDANCY OF IGNITERS FOR EACH GENERAL VICINITY.	E	13

TABLE 5.2-1: SUMMARY OF SPECIFIC IGNITER PLACEMENT AND DESIGN CRITERIA FOR SYSTEM 80+				PRIORITY
NO.	CRITERIA	BASIS	NOTES	
8.	ALL IGNITER PAIRS ARE POWERED VIA INDEPENDENT POWER SOURCES.	LOSS OF POWER TO ONE IGNITER SET WILL NOT COMPROMISE REGIONAL COVERAGE.	E	17
9.	IGNITERS LOCATED WITH REASONABLE EXPECTATIONS OF MAINTAINABILITY AND SURVEILLANCE.	TO ASSURE A FUNCTIONAL AND MAINTAINABLE SYSTEM IGNITERS ARE USUALLY LOCATED ON WALLS OR SURFACES ACCESSIBLE FOR SURVEILLANCE.	K	9
10.	NO MORE IGNITERS PLACED THAN REASONABLY NECESSARY.	SYSTEM 80+ IS DESIGNED WITH A LARGE CONTAINMENT VOLUME. THEREFORE, TO MEET THE GENERAL GUIDANCE OF CONTROLLING HYDROGEN TO BELOW 10 V/O GLOBAL, IT IS REQUIRED THAT ONLY 25% OF THE HYDROGEN PRODUCED BY A 100 % OXIDATION OF ZIRCALOY ACTIVE CLADDING BE REMOVED BY THE IGNITERS. SHERMAN-BERMAN ASSESSMENT OF DETONABILITY WITHIN SYSTEM 80+ IS LOW AND HYDROGEN CONTROL FOR PURPOSES OF PREVENTING DETONABILITY WOULD HAVE LIMITED RISK SIGNIFICANCE.		10
11.	LIMITED USE OF IGNITERS IN IRWST.	IGNITERS IN THE IRWST CAN PLAY A ROLE IN HYDROGEN CONTROL FOR THOSE UNUSUAL CIRCUMSTANCES WHEN THE HYDROGEN CONTENT IN THE IRWST IS COMBUSTIBLE. FOR MOST SEVERE ACCIDENT SITUATIONS THE IRWST HYDROGEN WILL BE NON-COMBUSTIBLE OR ONLY WEAKLY COMBUSTIBLE DUE TO THE LACK OF OXYGEN OR STEAM INERTING. THE NUMBER OF IGNITERS IN THE IRWST SHOULD BE LIMITED. THIS IS IMPORTANT SINCE MAINTAINABILITY AND TESTING IN IRWST DIFFICULT.		
12	ALL IGNITERS PLACED ABOUT 10 FEET BELOW A SOLID SURFACE (FLOOR, ETC.).	EXPERIMENTAL DATA INDICATES UPWARD BURNING TO INITIATE BETWEEN 4 AND 6 V/O H ₂ . LOCATION OF IGNITERS SEVERAL FEET BELOW SOLID FLOORS ENABLES MORE EFFICIENT USE OF THE IGNITER TO CONTROL CONCENTRATION AT THE LOWER END OF THE FLAMMABILITY RANGE. NTS DATA INDICATES COMBUSTION EFFICIENCY IS HIGHER AWAY FROM WALLS AND WHEN HYDROGEN CONCENTRATION IS BELOW 7 V/O. BASED ON NTS (VOLUME 75,000 FT ³) WITH ONE IGNITER LOCATED NEAR A WALL, COMBUSTION EFFICIENCY AT 7 V/O IS ON THE ORDER OF 30 TO 50 %).	H	6

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TABLE 5.2-1: SUMMARY OF SPECIFIC IGNITER PLACEMENT AND DESIGN CRITERIA FOR SYSTEM 80+				PRIORITY
NO.	CRITERIA	BASIS	NOTES	
13.	AVERAGE SUSTAINED CONCENTRATION OF CONTAINMENT HYDROGEN WITH A MINIMAL IGNITION SYSTEM BE LESS THAN 8 VOLUME PERCENT.	EXPERIMENTAL DATA ON HYDROGEN MIXING SUGGESTS THAT H ₂ CONCENTRATIONS IN VOLUMES ABOVE THE SYSTEM RELEASE POINT, WILL BE REASONABLY WELL MIXED TYPICALLY WITH A MAX/AVG CONCENTRATION GRADIENT OF UNDER 1.3.	J	14
14.	IGNITER SYSTEM SHOULD BE CAPABLE OF BURNING OFF SUFFICIENT HYDROGEN TO RENDER THE HYDROGEN CONCENTRATION IN THE CONTAINMENT ATMOSPHERE TO BELOW 10 V/O (GLOBAL AVERAGE) IN LESS THAN 4 HOURS FOLLOWING COMPLETE OXIDATION OF THE ZIRCALOY ACTIVE CLAD.	MAINTENANCE OF H ₂ CONCENTRATIONS BELOW 10 V/O ON A GLOBAL AVERAGE PROVIDES REASONABLE CONFIDENCE THAT A LOCAL DETONABLE MIXTURE IN A SEVERE ACCIDENT ENVIRONMENT (UNDER 13 V/O) WILL NOT OCCUR. THEREFORE, DETONATION THREATS TO CONTAINMENT WOULD BE UNLIKELY.	J	15
15.	NO IGNITERS NEAR PSV/SDS PIPING. ALL RELEASES FROM THE PSV/SDS CHANNEL INTO IRWST.	PRA DOES NOT CONSIDER A SIMULTANEOUS FAILURE OF THE PSV/SDS VALVES AND DOWNSTREAM PIPING. PROBABILITY OF ABOUT 10 ⁻⁸ PER YEAR		12
16.	WITH THE EXCEPTION OF THE DOME REGION IGNITERS IN THE DOMINANT FLOWPATHS SHOULD COVER A VOLUME OF LESS THAN 50,000 FT ³ .	NTS EXPERIMENTS INDICATE THAT A SINGLE IGNITER CAN EFFECTIVELY CONTROL HYDROGEN CONCENTRATION IN A 75,000 FT ³ SPHERE. THIS PLACEMENT CRITERIA IS WITHIN THE EXPERIMENTAL DATA BASE.		5
17.	IGNITERS TO BE PLACED AT POSITIONS ASSOCIATED WITH SMALLER SECONDARY FLOW PATTERNS.	THIS PLACEMENT PROVIDES COVERAGE FOR FLOW AREAS WHICH ARE NOT EXPECTED TO BE HYDROGEN RICH. HOWEVER, THEIR PRESENCE WILL CONTRIBUTE TO INCREASED HYDROGEN COMBUSTION AND ASSURE ALL POTENTIAL FLOWPATHS WILL HAVE IGNITER CAPABILITY REGARDLESS OF ANTICIPATED FLOWPATHS.	D	8
18.	MULTIPLE LEVELS OF BURNING IN SECONDARY FLOW PATHS.	INCREASES EFFECTIVENESS OF IGNITERS THAT BURN AT LOW CONCENTRATIONS		18

NOTES ON TABLE 5.2-1

A	<p>THIS CRITERIA IS INTERPRETED AS REQUIRING THE PLACEMENT OF IGNITERS ABOVE EACH OF THE 2 HOT LEGS, 4 RCP DISCHARGE LEGS AND 4 RCP SUCTION LEGS AND PRESSURIZER SURGE LINE.</p> <p>ADDITIONAL PLACEMENT MAY ALSO BE REQUIRED IN VICINITY OF DVI LINES, CHARGING AND LETDOWN, SHUTDOWN COOLING, DVI AND SIT LINES. ONLY LINE SIZES SUFFICIENT TO BE CONSIDERED A SMALL LOCA CONSIDERED.</p> <p>FOR PSV /SDS OPERATION HYDROGEN SOURCE TO THE CONTAINMENT WILL BE LOCATED AT IRWST VENTS (ABOVE GROUND FLOOR 91-9" ELEVATION AND IN VICINITY OF OVERFLOW PIPE IN HOLDUP VOLUME).</p> <p>POST-VB HYDROGEN SOURCE TO THE CONTAINMENT IS VIA THE VESSEL BREACH SCENARIO. THIS INTRODUCES HYDROGEN INTO THE REACTOR CAVITY. THUS, IGNITERS WILL BE LOCATED IN REACTOR CAVITY AND AT THE DOMINANT EXITS OF THE REACTOR CAVITY (ASSUMING FLOODED CAVITY). (1 REQUIRED; 3 IGNITERS PER LOCATION SELECTED FOR BACKUP).</p>
B	<p>BECAUSE OF CLOSE PROXIMITY OF PRIMARY COOLANT PIPING AND SAFETY/CONTROL LINES, ONE LOCATION MAY BE COVERED BY MORE THAN ONE IGNITER.</p>
C	<p>SYSTEM 80+ HAS SEVERAL VENTED COMPARTMENTS. THESE INCLUDE THE RHR HX ROOM AND LETDOWN HX ROOM. SINCE THESE ROOMS ARE LOCATED LOW IN THE CONTAINMENT AND ARE NOT IN DIRECT PROXIMITY TO A HYDROGEN SOURCE THERE IS NO RECOMMENDATION FOR IGNITER PLACEMENT.</p> <p>THE PRESSURIZER HOUSING REPRESENTS A VENTED TUNNEL WITH MORE THAN 100 FT² AREA VENT LOCATED TOWARDS THE TOP OF THE COMPARTMENT. USING CRITERION 2 AN IGNITER WILL BE LOCATED AT THE EXITS OF THE HOUSING AND THEREFORE ADDITIONAL IGNITERS WITHIN THE HOUSING IS NOT NECESSARY.</p> <p>FOR POST VB RELEASE THE HYDROGEN RELEASED FOLLOWING VB IS VENTED TO THE LOWER COMPARTMENT VIA THE CAVITY COOLING VENTILATION ROOM. THIS ROOM IS WELL VENTED. HOWEVER, AN IGNITER WILL BE PLACED IN THIS AREA BECAUSE IT REPRESENTS A HYDROGEN SOURCE TO THE CONTAINMENT VIA CRITERION 2.</p> <p>A SMALL REGION OF CONTAINMENT ADJACENT TO THE PRESSURIZER HOUSING APPEARS POORLY VENTED AND SHOULD BE FITTED WITH IGNITERS DUE TO CRITERION 3.</p> <p>IN ORDER TO ENSURE A DOMINANT CHIMNEY FLOW PATTERN THE HVAC HEADER IS POORLY VENTED (LEAVING THE LOW RESISTANCE PATHWAY THROUGH THE RCS GRATED REGIONS). THUS, IGNITERS WILL BE LOCATED IN THE HVAC HEADER REGION.</p>

NOTES ON TABLE 5.2-1	
D	THESE IGNITERS ARE NOT PART OF MINIMAL SET AND NEED NOT HAVE BATTERY BACKUP.
E	PAIRS LOCATED IN THE SAME GENERAL REGION, BUT MAY BE SEPARATED BY DISTANCE.
F	A MINIMAL IGNITER SET WILL BE POWERED OFF BATTERIES, COMBUSTION TURBINE (CT), DIESEL GENERATOR (DG) & OFFSITE SOURCE; A SUPPLEMENTAL IGNITER SET WILL BE POWERED OFF OF CT, DG & OFFSITE SOURCE.
G	MULTILEVEL BURNING IS PARTICULARLY IMPORTANT TO KEEP LOWER HYDROGEN CONCENTRATIONS WHERE COMBUSTION COMPLETENESS IS EXPECTED TO BE LOW.
H	NTS DATA INDICATES HYDROGEN CONTROL IN LARGE OPEN VOLUMES CAN BE PERFORMED WITH A SINGLE IGNITER. IGNITER LOCATION HAS SOME IMPACT ON COMBUSTION COMPLETENESS AND FLAMMABILITY LIMITS.
I	IF NECESSARY ADDITIONAL IGNITERS WILL BE ADDED TO COVER BREAKS IN RCS CONNECTING PIPING
J	MINIMAL IGNITER SET INCLUDES 33 IGNITERS (JULY 6, 1993 DESI RECOMMENDATION).
K	AN EXCEPTION TO THIS RULE IS EXPECTED TO BE THE DOME IGNITERS WHICH ARE TO BE SUSPENDED 10 TO 15 FEET BELOW THE DOME INNER SURFACE.

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6.0 DESCRIPTION OF THE SYSTEM 80+ HYDROGEN MITIGATION SYSTEM

This section provides the details with respect to the Hydrogen Mitigation System (HMS) igniter design, placement and operation. The HMS is designed to accommodate the hydrogen production from 100% active fuel clad metal-water reaction and limit the containment average hydrogen concentration to 10 % in accordance with 10 CFR 50.34(f) for a severe accident. A detailed description of the System 80+ HMS is provided in Section 6.2.5.

6.1 IGNITER PLACEMENT

The System 80+ containment utilizes eighty (80) shielded GMAC model 7G thermal igniter glow plug controlling hydrogen in roughly forty containment locations. This igniter glow plug has been extensively tested, and is incorporated in the igniter system for the GE Mark III containment and several Westinghouse PWRs with ice condenser containments. Each igniter is powered by a 120/14 V step-down transformer designed to provide a minimum surface temperature of 1700°F. The igniter assembly is illustrated in Figure 6.1-1.

The System 80+ HMS employs 80 igniters, providing coverage for about 40 containment regions. This level of igniter coverage is typically consistent with the 90 employed in the Grand Gulf design and the 68 established for the Sequoyah ice condenser PWR. Hypothetical igniter systems design for Zion and Surry large dry PWRs included 76 igniters. Table 6.1-1 summarizes the System 80+ igniter placement. Detailed three dimensional placement drawings for these igniters are presented in Figures 6.1-2 through 6.1-6.

6.2 IGNITER POWER SUPPLY

In order to assure high igniter availability, the igniters are powered from several independent Class 1E power sources. All eighty igniters may be powered via (1) offsite power source, (2) emergency diesel generators, and (3) alternate AC power source (combustion turbine generator). A minimum of 32 igniters can be powered from station batteries (16 each via each emergency bus as identified in Table 6.1-1. The station batteries can provide four (4) hours of power for operation of these igniters. As discussed in Section 7.0, MAAf analyses suggest that approximately two (2) hours of igniter operation is sufficient to burn off enough hydrogen to limit the global containment hydrogen concentration to below 10 volume percent for a 100% active fuel clad oxidation severe accident scenario.

6.3 IRWST VENT SYSTEM

In addition to the igniter placement and power supply, another essential feature of the HMS is the powered shut IRWST vent system. This system is used to aid in the removal of hydrogen from the IRWST so that the hydrogen passing through the tank may be burned elsewhere. The IRWST vent system consists of several vents connecting the IRWST with the lower containment.

Four vents will be located within the steam generator wing walls. The remaining vents will be distributed about the containment. These vents occupy a total vent floor area of about 600 ft² with a free flow area of 400 ft². The difference between these two areas is the dense grating which is required for floor structural integrity and protection of the sump.

The 400 ft² vent area is considered sufficient to maintain the hydrogen concentration in the IRWST freeboard space to about 10 volume percent. The steam and oxygen concentrations within the IRWST would be such that even at higher hydrogen concentrations, the steam, air and hydrogen mixture is not likely to be detonable.

The selection of the 400 ft² IRWST vent area is based on maximizing the venting capability of the IRWST while simultaneously minimizing the adverse impact of larger vents on plant safety and operational aspects. Specific factors that were of concern in limiting the vent size include:

(1) Cleanliness and Boration of IRWST Water

Leakage through the vents and/or failure of the vent to remain closed can cause introduction of contaminants/debris and unbored water into the IRWST. In addition, moisture escaping from the IRWST through the vents can result in a corrosive atmosphere in the containment. These effects can compromise the effectiveness of the safety injection function and/or result in plant equipment damage.

(2) Impact on Plant Operation/Maintenance

Since all containment flows are currently required to enter the Holdup Volume Tank for boration, the floor vents would require raised ledges to prevent leakage into the IRWST. These protrusions could potentially obstruct free movement throughout the containment and limit equipment laydown areas available for maintenance operations.

(3) Reduction in Structural Capability

The structural strength of the IRWST top lid can be impacted due to the number and size of the IRWST vents. Larger vent area could adversely impact the structural strength of the lid.

As an alternative to the large IRWST venting described above, a combination of a smaller IRWST vent area and Passive Autocatalytic Recombiners (PARs) in the vicinity of the vents were considered as an optional design feature to maintain the hydrogen concentration within the IRWST below 10%.

The PARs would recombine the hydrogen and oxygen passively without combustion. In this exothermic process heat generated within the PAR would set up a density difference across the PAR to initiate significant natural circulation flows near the vicinity of the PAR. In the proposed alternate option, a number of PARs (about 4) are located near the vicinity of the IRWST vents which have a total area of about 150 ft². These vents will be distributed in four locations along the top lid of the IRWST. As

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in the first option, the vents will powered shut with manual actions or a loss of power causing their opening. As the PARs begin to recombine hydrogen and oxygen, sufficient natural circulation flow between the IRWST freeboard space and the lower compartment of the containment will be initiated. This flow will be adequate to maintain the IRWST hydrogen concentration below 10%.

6.4 IGNITER ACCESSIBILITY AND MAINTENANCE

All hydrogen igniters have been evaluated for required access for testing and maintenance. The most inaccessible igniter locations are in the containment dome. These igniters can be reached from the polar crane with a temporary scaffolding / ladder arrangement. Igniters in the steam generator cubicles are placed to allow access from existing platforms. Igniters in the IRWST utilize tubes in the IRWST roof to allow the igniter assembly to be retracted to the 91+9 elevation for testing and maintenance without requiring IRWST entry. Igniters are generally located 7 to 10 feet above floors to allow easy access without impeding personnel passage or becoming a personnel safety hazard.

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TABLE 6.1-1
SYSTEM 80+ HYDROGEN IGNITER LOCATIONS

TAG NO.	LOCATION	ELEVATION (Centerline)	AZIMUTH (Degrees)	RADIUS (Ft. & in)
1A*	REACTOR CAVITY ICI AREA	70'+0"	22	13'-0"
1B*	REACTOR CAVITY ICI AREA	70'+0"	314	13'-1"
2A*	IRWST AREA	88'+6"	44	44'-11"
2B*	IRWST AREA	88'+6"	136	44'-11"
3A*	IRWST AREA	88'+6"	224	44'-11"
3B*	IRWST AREA	88'+6"	316	44'-11"
4A	MAVEC AREA	89'+6"	355	28'-2"
4B*	MAVEC AREA	89'+6"	351	35'-8"
5A*	IRWST VENT OUTLET	100'+0"	16	43'-7"
5B	IRWST VENT OUTLET	100'+0"	342	44'-1"
6A	IRWST VENT OUTLET	99'+0"	154	42'-6"
6B*	IRWST VENT OUTLET	99'+0"	206	42'-6"
7A*	EL. 91'+9" STEAM GEN. 2 WING WALL	100'+0"	59	48'-2"
7B	EL. 91'+9" STEAM GEN. 2 WING WALL	100'+0"	73	52'-11"
8A	EL. 91'+9" STEAM GEN. 2 WING WALL	100'+0"	108	52'-7"
8B*	EL. 91'+9" STEAM GEN. 2 WING WALL	100'+0"	130	46'-11"

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TABLE 6.1-1
SYSTEM 80+ HYDROGEN IGNITER LOCATIONS

TAG NO.	LOCATION	ELEVATION (Centerline)	AZIMUTH (Degrees)	RADIUS (Ft. & in)
9A*	EL. 91'+9" STEAM GEN. 1 WING WALL	100'+0"	224	44'-11"
9B	EL. 91'+9" STEAM GEN. 1 WING WALL	100'+0"	252	52'-7"
10A	EL. 91'+9" STEAM GEN. 1 WING WALL	100'+0"	287	52'-11"
10B*	EL. 91'+9" STEAM GEN. 1 WING WALL	100'+0"	314	46'-9"
11A	LETDOWN HEAT HX RM.	100'+0"	151	64'-8"
11B	LETDOWN HEAT HX RM.	100'+0"	178	56'-1"
12A	REGEN. HX RM.	100'+0"	199	59'-1"
12B	REGEN. HX RM.	100'+0"	206	62'-8"
13A*	EL. 91+9 HVAC DIST. HEADER	105'+0"	30	82'-2"
13B	EL. 91+9 HVAC DIST. HEADER	105'+0"	75	82'-2"
14A	EL. 91+9 HVAC DIST. HEADER	105'+0"	120	82'-2"
14B*	EL. 91+9 HVAC DIST. HEADER	105'+0"	165	82'-2"
15A*	EL. 91+9 HVAC DIST. HEADER	105'+0"	210	82'-2"
15B	EL. 91+9 HVAC DIST. HEADER	105'+0"	255	82'-2"
16A	EL. 91+9 HVAC DIST. HEADER	105'+0"	300	82'-2"
16B*	EL. 91+9 HVAC DIST. HEADER	105'+0"	345	82'-2"

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TABLE 6.1-1
SYSTEM 80+ HYDROGEN IGNITER LOCATIONS

TAG NO.	LOCATION	ELEVATION (Centerline)	AZIMUTH (Degrees)	RADIUS (Ft. & in)
17A*	L. 115+6 O.D. CRANE WALL	125'+0'	25	69'-4"
17B	EL. 115+6 O.D. CRANE WALL	125'+0"	66	69'-4"
18A	EL. 115+6 O.D. CRANE WALL	125'+0"	114	69'-4"
18B	EL. 115+6 O.D. CRANE WALL	125'+0"	152	69'-4"
19A	EL. 115+6 O.D. CRANE WALL	125'+0"	205	69'-4"
19B*	EL. 115+6 O.D. CRANE WALL	125'+0"	246	69'-4"
20A	EL. 115+6 O.D. CRANE WALL	125'+0"	294	69'-4"
20B	EL. 115+6 O.D. CRANE WALL	125'+0"	335	69'-4"
21A	EL. 115+6 GRATING HATCH AREA	123'+6"	346	43'-1"
21B	EL. 115+6 GRATING HATCH AREA	123'+6"	13	43'-1"
22A*	STEAM GEN. 2 AREA	126'+3"	66	46'-3"
22B	STEAM GEN. 2 AREA	126'+3"	114	46'-3"
23A	STEAM GEN. 2 AREA	126'+3"	54	23'-6"

TABLE 6.1-1
SYSTEM 80+ HYDROGEN IGNITER LOCATIONS

TAG NO.	LOCATION	ELEVATION (Centerline)	AZIMUTH (Degrees)	RADIUS (Ft. & in)
23B*	STEAM GEN. 2 AREA	126'+3"	127	23'-6"
24A*	STEAM GEN. 1 AREA	126'+3"	294	46'-3"
24B	STEAM GEN. 1 AREA	126'+3"	247	46'-3"
25A	STEAM GEN. 1 AREA	126'+3"	307	23'-6"
25B*	STEAM GEN. 1 AREA	126'+3"	234	23'-6"
26A*	STEAM GEN. 2 AREA	164'+0"	66	46'-3"
26B	STEAM GEN. 2 AREA	164'+0"	114	46'-3"
27A	STEAM GEN. 2 AREA	164'+0"	43	28'-7"
27B*	STEAM GEN. 2 AREA	164'+0"	137	28'-7"
28A*	STEAM GEN. 1 AREA	164'+0"	294	46'-3"
28B	STEAM GEN. 1 AREA	164'+0"	246	46'-3"
29A	STEAM GEN. 1 AREA	164'+0"	317	28'-7"
29B*	STEAM GEN. 1 AREA	164'+0"	223	28'-7"
30A*	REFUEL CAVITY	154'+0"	45	22'-6"
30B	REFUEL CAVITY	154'+0"	135	22'-6"
31A	REFUEL CAVITY	154'+0"	315	22'-6"
31B*	REFUEL CAVITY	154'+0"	225	22'-6"
32A	PRESSURIZER	188'+10"	227	46'-9"
32B	PRESSURIZER	188'+10"	248	50'-9"

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TABLE 6.1-1
SYSTEM 80+ HYDROGEN IGNITER LOCATIONS

TAG NO.	LOCATION	ELEVATION (Centerline)	AZIMUTH (Degrees)	RADIUS (Ft. & in)
33A	PRESSURIZER	188'+10"	216	64'-8"
33B	PRESSURIZER	188'+10"	252	64'-8"
34A*	EL. 146 I.D. CRANE WALL	200'+0"	40	64'-8"
34B	EL. 146 I.D. CRANE WALL	200'+0"	77	64'-8"
35A	EL. 146 I.D. CRANE WALL	200'+0"	125	64'-8"
35B*	EL. 146 I.D. CRANE WALL	200'+0"	175	64'-8"
36A	EL. 146 I.D. CRANE WALL	200'+0"	310	64'-8"
36B	EL. 146 I.D. CRANE WALL	200'+0"	355	64'-8"
37A	CONT. DOME	237'+0"	0	56'-7"
37B	CONT. DOME	237'+0"	45	56'-7"
38A*	CONT. DOME	237'+0"	90	56'-7"
38B	CONT. DOME	237'+0"	135	56'-7"
39A	CONT. DOME	237'+0"	180	56'-7"
39B	CONT. DOME	237'+0"	225	56'-7"
40A	CONT. DOME	237'+0"	270	56'-7"
40B*	CONT. DOME	237'+0"	315	56'-7"

* denotes igniters that can be powered by Class 1E batteries, in addition to offsite power source, diesel generators, or alternate AC combustion turbine generator

[illegible]

1. WELD TO GROUND THE TRANSDUCER CABLE TERMINAL LEAD TO THE PANEL MOUNTING STUD AND CONNECT A MICROPHONE CABLE TERMINAL LEAD TO THE PANEL SUPPLY M1; WELD ALL WOUND LEADS TO BE WOUND TO THE GROUND PLUG TO 2 PLY TO CENTER AND WOUND GLOW PLUG ON RIGHT SIDE OF BOX.

2. WELD TO SPACE THE TRANSDUCER 115V MINIMUM LEADS AND THE INPUT WOUND LEADS AS SHOWN IN AN APPROVED WIRING USING SPACE CONNECTIONS. ALL CABLES BEHIND COVER.

3. WELD TO SPACE THE CABLES AS SHOWN IN THE INTERCONNECT SCHEDULES. WELD TO SPACE CABLES TO THE CABLE CONDUIT WIRING CABLE (SEE INTERCONNECT SCHEDULES).

4. WELD TO WOUND TRANSDUCER AS SHOWN.

5. ALL CABLES HAVE ONE SHIELD CONDUCTOR WHICH IS NOT SPECIFIED IN INTERCONNECT SCHEDULES.

6. WELD TO SPACE SHORT TO GROUND PIECES OF WIRE TO THE 10 GAUGE CABLE CONNECTIONS TO FACILITATE CONNECTIONS TO TRANSDUCER PINS. P 2 CABLES ARE SHOWN THEY ARE TO BE JOINED AT THIS POINT AND WELDED TO THE 12 GAUGE CONNECTION WIRE (TERMINAL B&I) SHOWN AS 3 PLY (TERMINAL A SH&I, 1/2 INCH) & 500L (TERMINAL C SH&I) THROUGH A 3/16 INCH DIA. (TERMINAL B&I) SHOWN AS 3 PLY & 500L AS SHOWN ON DWG.

7. WELD TO SPACE THE CABLES AS SHOWN IN THE INTERCONNECT SCHEDULES. WELD TO SPACE CABLES TO THE CABLE CONDUIT WIRING CABLE (SEE INTERCONNECT SCHEDULES).

8. WELD TO SPACE THE CABLES CONDUIT WIRING AND TRANSDUCER PINS.

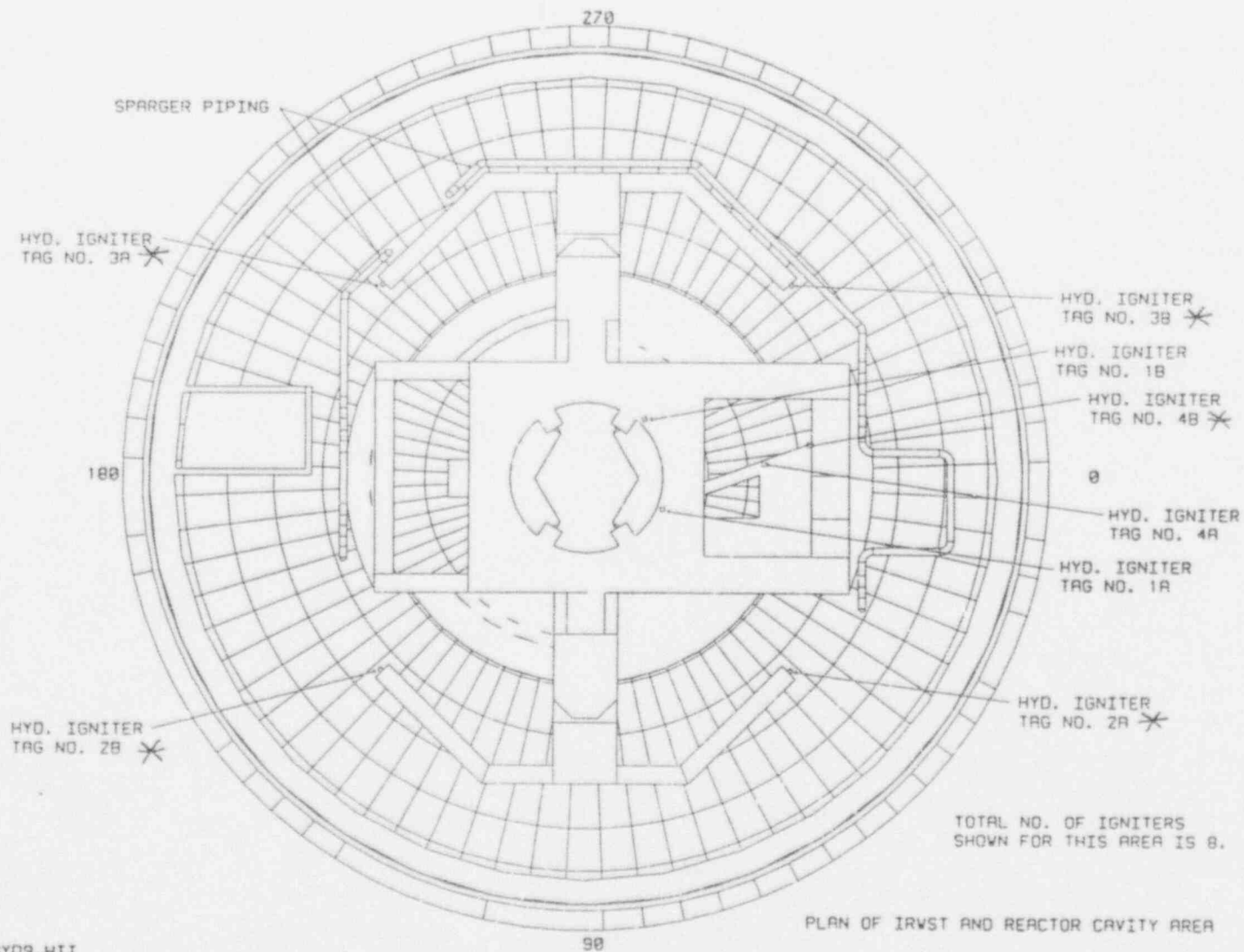
9. WOUND (PINS C&I) TO THE SPECIAL CABLES SPECIFIED IN NOTE 12 OF THE SUPPLY ACTION BOX LOCATED OUTSIDE THE END WALL OF THE ICE CONDUIT.

10. CABLES SHOWN DOWNSIDE OF THE SUPPLY ACTION BOX WELDED TO THE SUPPLY ACTION BOX. WELD TO SPACE THE CABLES THROUGH THE END WALL OF THE ICE CONDUIT AND THE L.P. SEGMENT OF TERMINAL 1 WOUND AS PER INSTALL IN ALLED AND WELDED.

11. WELD TO SPACE THE CABLES (SEE INTERCONNECT SCHEDULES) AT ALL TERMINALS.

K-40

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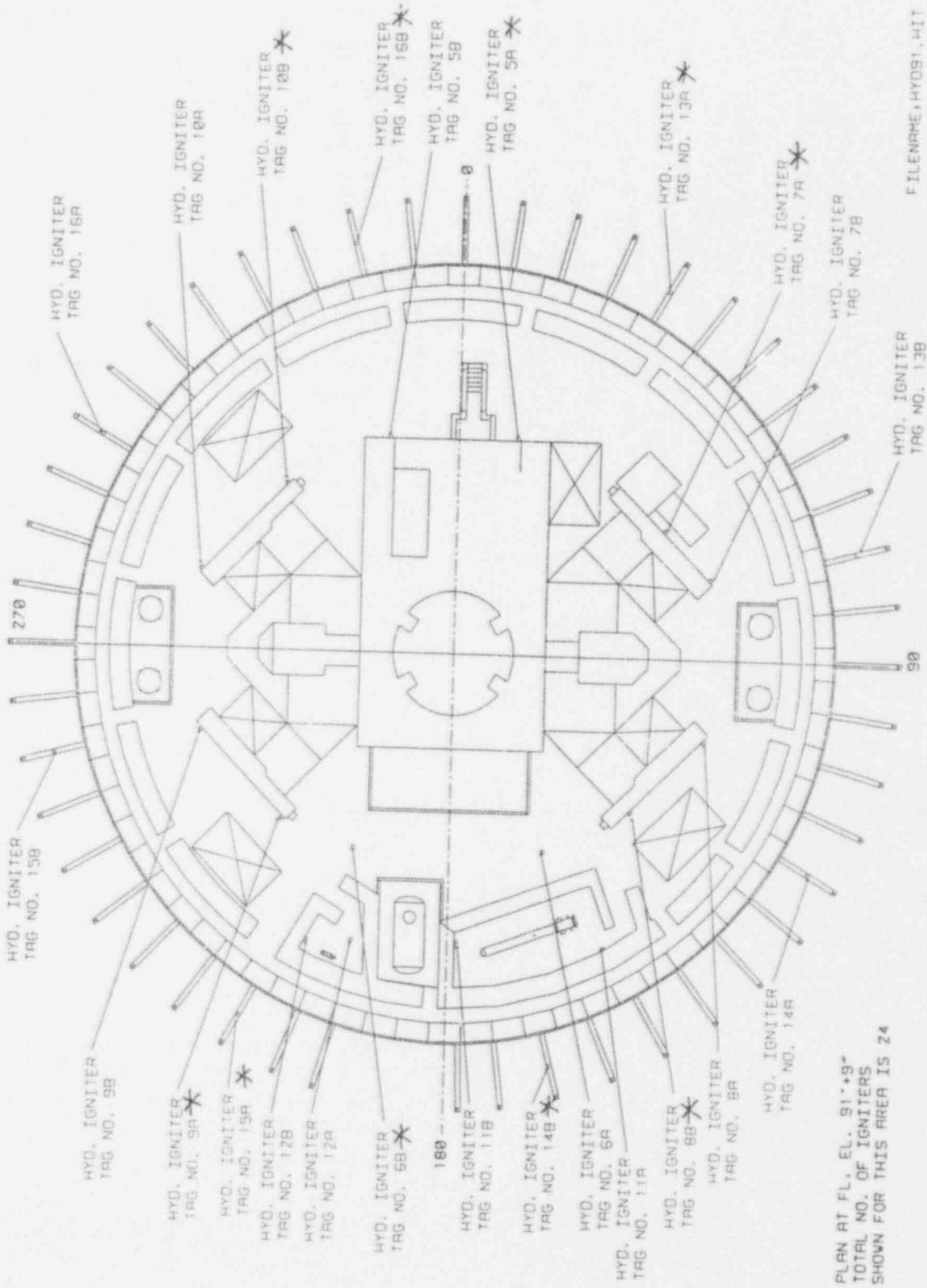
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FIGURE 6.1-2: SYSTEM 80+ HYDROGEN IGNITER LOCATION: PLAN VIEW OF IRWST AND CAVITY AREA
(IGNITER PAIRS 1,2,3 AND 4)

K-41

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FILE NAME: HYD091.DIT

FIGURE 6.1-3: SYSTEM 80+ HYDROGEN IGNITER LOCATION: PLAN VIEW AT ELEVATION 91'+9"
(IGNITER PAIRS 5,6,7,8,9,10,11,12,13,14,15,16)

PLAN AT FL. EL. 91'+9"
TOTAL NO. OF IGNITERS
SHOWN FOR THIS AREA IS 24

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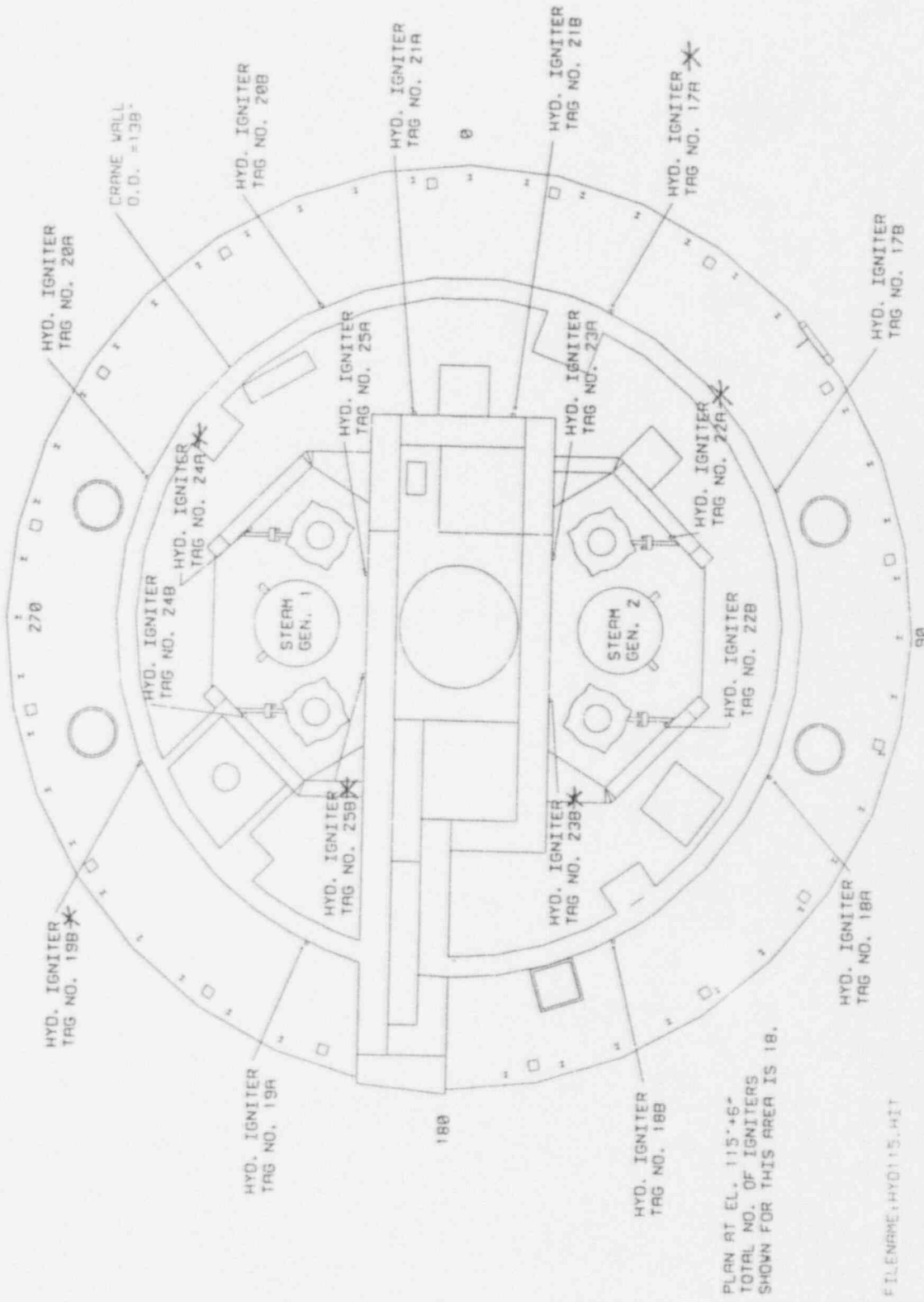


FIGURE 6.1-4: SYSTEM 80+ HYDROGEN IGNITER LOCATION: PLAN VIEW AT ELEV. 115'+6"
(IGNITER PAIRS 17,18,19,20,21,22,23,24,25)

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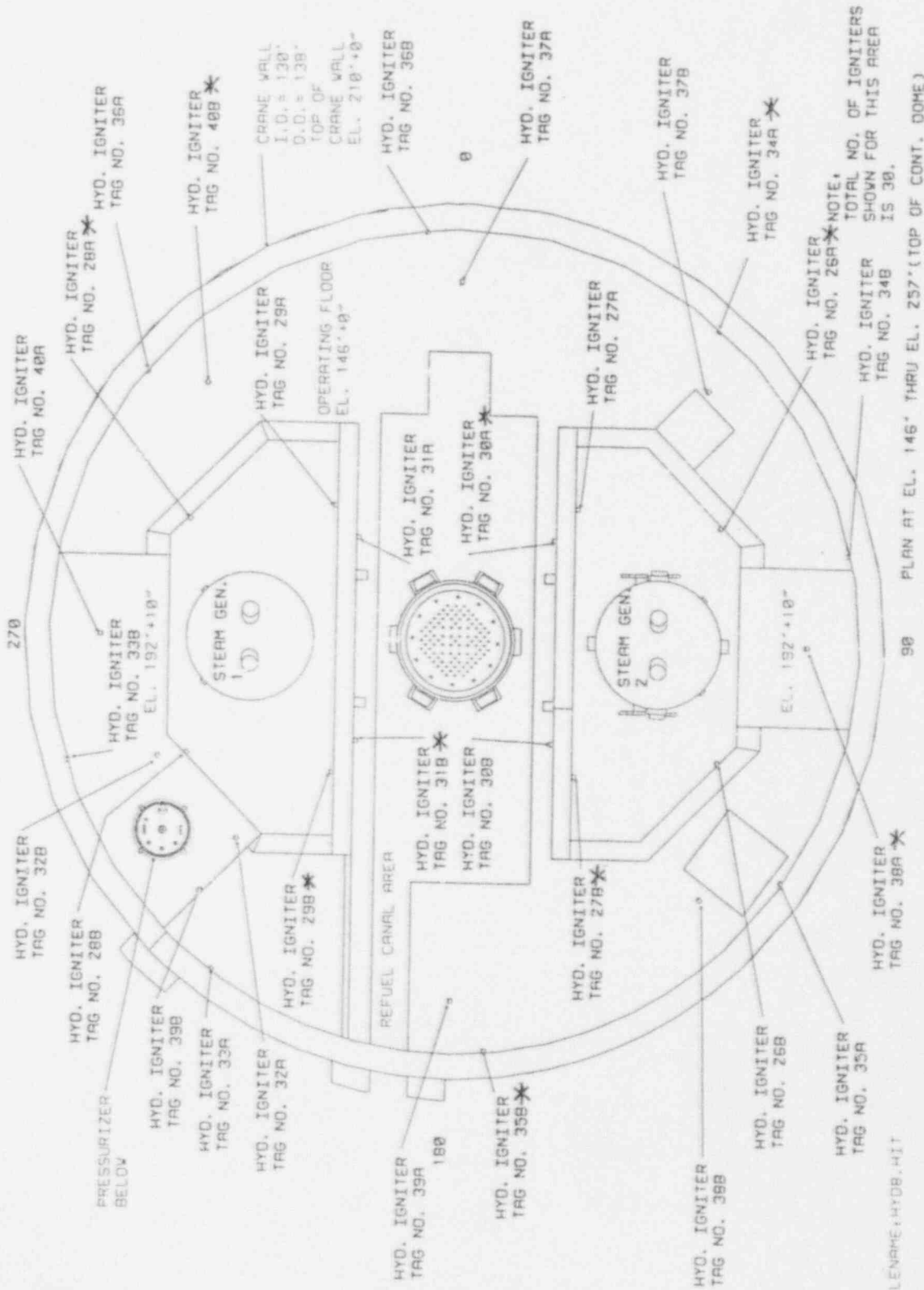
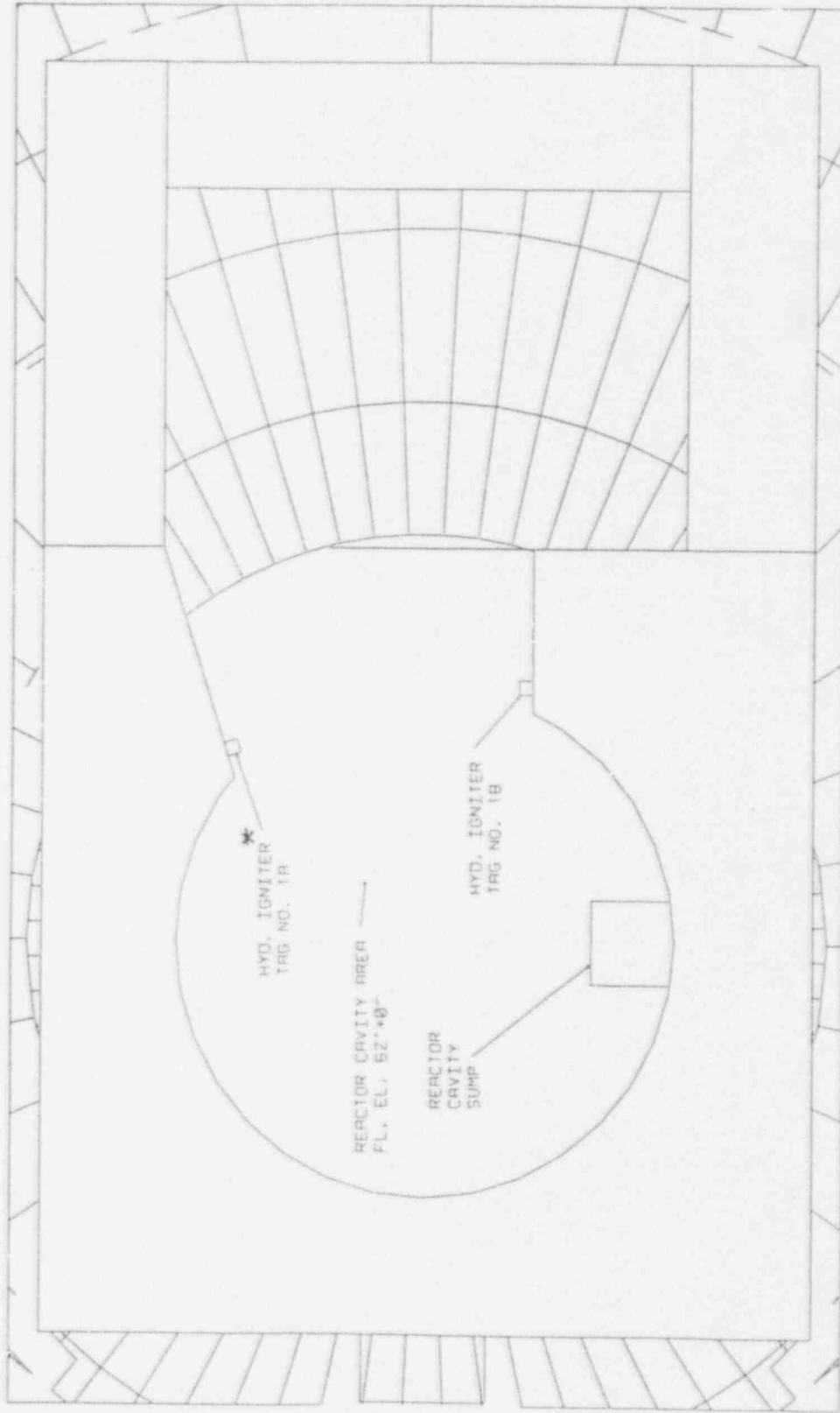


FIGURE 6.1-5: SYSTEM 80+ HYDROGEN IGNITER LOCATION: PLAN VIEW AT ELEV. 146' THROUGH 247' (TOP OF DOME) - IGNITER PAIRS 26 THROUGH 40)

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PARTIAL PLAN REACTOR CAVITY AREA

FIGURE 6.1-6: SYSTEM 80+ HYDROGEN IGNITER LOCATION: PLAN VIEW OF REACTOR CAVITY (IGNITER PAIRS DETAIL VIEW)

K-45

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7.0 ANALYTICAL VERIFICATION OF THE SYSTEM 80+ IGNITER SYSTEM

This section summarizes the highlights of the System 80+ MAAP 4 containment analyses performed to assess the System 80+ HIS. The MAAP 4 code was used to study hydrogen mixing and combustion in the System 80+ containment. The key goals of the study were to assess the potential for hydrogen build-up in the containment and to calculate the best-estimate response of the hydrogen igniter system.

MAAP 4 contains a state-of-the-art lumped parameter model for containment thermal-hydraulics. The model was specially constructed to model natural circulation in advanced light water reactor containments. A complete description of the code is available in Reference 28. The key aspects of the model which bear on its use in the hydrogen calculations for system 80+ are as follows:

- a. Mechanistic, semi-implicit models for gas, water, and energy transport between control volumes
- b. Models for both unidirectional and counter-current flow²⁹ through containment junctions
- c. Stable treatment of water-solid regions; these can develop in System 80+ calculations if the IRWST pool is sub-nodalized (this was done here) or if the cavity flooding system is activated
- d. Flexible modelling of containment heat sinks
- e. Advanced modelling of hydrogen combustion. Both non-global burns initiated by the hydrogen igniters and global burns are treated using a single, unified framework. This model has been successfully compared to a great variety of experiments³⁰.

7.1 CONSTRUCTION OF PLANT MODEL

A detailed (23 control volumes, 35 junctions, and 37 heat sinks) containment model was constructed as shown in Figures 7.1-1 and 7.1-2. Considerable effort was taken to minimize artificial mixing which can be caused by the limitations inherent in lumped parameter containment codes.

Calculations were performed using the number and location of the igniters presented in Table 6.1-1. In addition the MAAP 4 simulations employed an IRWST vent area of 400 ft².

A recent international standard problem (ISP-29) tested the ability of various lumped parameter codes to predict hydrogen concentrations in the HDR containment during experiment E11.2. The results indicated that all the codes tended to over-predict mixing³¹; whereas, very little hydrogen was measured below the elevation at which the hydrogen was injected, the codes predicted substantial mixing.

Part of this tendency to over-predict mixing is a consequence of inherent assumptions used in lumped parameter codes, i.e. the fact that control

volumes are assumed to be well-mixed can lead to numerical diffusion. However, these problems can at least be minimized by careful construction of the containment model. For example, the effects of numerical diffusion were reduced in this study by employing a relatively large number of nodes.

More important, in the original System 80+ model it was observed that a null transient (no source of mass or energy to the containment) resulted in persistent gas flow rates on the order of 1 kg/sec or more. This was found to be caused by the assumption of uniform density in each node if the node boundaries of inter-connected nodes were defined in such a way that they overlapped. By adjusting the control volume boundaries and junction elevations slightly, it was found possible to reduce these "phantom flows" to less than 1 gram/second during null transients.

To confirm the success of the renodalization effort, a special case of the System 80+ model was created that eliminated reactor vessel convective heat transfer to the reactor cavity and which established a large return flow path from the refuelling pool area to the lower compartment region. The former was done to eliminate (physically reasonable) gas flows through the cavity which are caused by the "chimney" effect that vessel heating creates in the cavity. The second change allows the large hydrostatic heads which develop between the steam generator compartments and the upper compartment and refueling pool area (caused by convective heating and the introduction of hydrogen) to cause return flows which do not involve the lower cavity region. Both changes were intended to mimic the HDR situation. For this case, a high degree of stratification was calculated to be maintained between the bulk of containment (above the hydrogen injection point) and the lower reactor cavity (which lies below). This is analogous to the behavior observed in experiment E11.2 and affords added confidence in the results of the MAAP 4 calculations.

The use of a large number of control volumes also allowed the igniter placement relative to the hydrogen igniter points to be represented in a detailed fashion. More detail on the containment model can be found in Reference 31.

7.2 METHODOLOGY

A special version of MAAP 3.0B, which modelled features specific to the System 80+ design, was used in the PRA to calculate primary system and containment response during severe accidents (see section 19.11.5 of CESSAR-DC). To avoid the need to modify MAAP 4 to represent the special features of the System 80+ primary system, hydrogen and steam flow rates, and energy transfer rates from the primary system were calculated with MAAP 3.0B and these were then fed into the MAAP 4 containment model. As such, only MAAP 4's containment model was used for this work.

The procedure used to perform a calculation consisted of several steps:

1. A calculation was made using the standard MAAP 3.0B model for System 80+. Steam and hydrogen flow to the containment as well as convective energy transfer between RCS heat sinks and the containment were stored in a file as functions of time.

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2. For the purpose of these calculations, it was desired to introduce a mass of hydrogen to containment equivalent to reaching 100 percent of the active cladding (about 2400 lbm). To generate the full 2400 lbm of hydrogen in the core, it was necessary to increase the MAAP-predicted hydrogen generation by a factor between 1.5 and 1.75, depending on the accident sequence. To accomplish this in a manner which would not unduly distort the containment response, the period of core damage was extended by adding a "tail" to the calculated hydrogen release curve. The hydrogen release rate during this period was set equal to the average MAAP-calculated hydrogen release over the period of core damage (~ 0.2 kg/sec), and the length of the "tail" was determined by the total additional hydrogen mass that was needed to bring the in-vessel oxidation to 100 percent of the active cladding. No steam was released during this interval, i.e. the reaction was assumed to be steam-limited during this time, and vessel failure was assumed to occur at the end of this extension.
3. These quantities were then fed into the MAAP 4 containment model. Cases with igniters operational and igniters disabled were considered separately. Hydrogen combustion in control volumes not containing igniters was suppressed; this is quite conservative with respect to hydrogen concentrations, since combustion initiating at an igniter could easily propagate into horizontally-adjacent or higher nodes bearing hydrogen concentrations in excess of about 6 percent³⁰.
4. At vessel failure, the entire debris mass was released to the cavity (node 1) over a 10 second interval. The hydrogen, steam, and water present in the RCS at vessel failure in the MAAP 3.0B calculations were released into the cavity over a 30 second interval. The intent was to capture approximately the impact of the blowdown on the steam concentrations. Finally, the portion of the accumulator water that was still in the accumulators at vessel failure in the MAAP 3.0B calculation was released to the cavity over a 60 second interval.

The two representative accident sequences which were studied were a station blackout (SBO) with activation of the safety depressurization system and a small break LOCA (SBLOCA) with no safety injection and no depressurization system activation. Containment spray and igniter availability were varied in all three sequences. That is, some of the sequences are "SBO-like" insofar as the primary system is concerned, but containment sprays and igniters may still be available. Debris dispersal was not modelled in these hydrogen distribution calculations, and activation of the flooders was not considered.

7.3 RESULTS OF CALCULATIONS

Several calculations were performed. To some degree, this reflects changes in the IRWST vent area and location and the number and location

DRAFT

of igniters that occurred while the analyses were being performed. A variety of sensitivity calculations were also run.

A few of the key calculations will be summarized in the subsections below. Key results of the base cases are presented in Tables 7.1-1 and 7.1-2.

7.3.1 Model Verification Cases

As discussed above, two cases were run to verify the containment nodalization. The first was the null transient. This case was designed to show that persistent flow would not develop in the absence of heat or mass addition to containment. All of the containment nodes were initialized to the same pressure, temperature, and humidity. The heat sinks were also initialized to the same temperature. Also, the IRWST was initialized without water to prevent evaporation from driving flow. As expected for this case, the containment quickly reached equilibrium, and gas flow was stopped.

The second case was an attempt to approximately simulate HDR-like experimental conditions in System 80+ to demonstrate the ability of the model to predict global stratification. As discussed previously, in this case there was no convective heat input to containment, and the hydrogen/steam source was introduced in a node above the cavity. The upper portion of containment was calculated to be fairly well mixed, while the cavity contained a much lower hydrogen concentration.

7.3.2 Small LOCA Cases

In these cases the primary system behaved as it would during a small break LOCA. Steam and hydrogen were released continuously to the lower steam generator area via a broken pipe. At vessel failure, a small mass of remaining hydrogen in the reactor vessel was released to the reactor cavity. The total hydrogen released was equivalent to reacting 100% of the active fuel cladding.

1. In the case without igniters available, there were no hydrogen burns. Prior to vessel failure, the hydrogen concentration in the bulk of containment was about 10%. In the node that contained the LOCA, a hydrogen concentration peak of 11% was observed. The IRWST behaved similar to the bulk of containment. There were no hydrogen concentration spikes at vessel failure. After vessel failure, the containment mixed fairly well to obtain a 8% hydrogen concentration everywhere.
2. In the case with igniters available, a total of 500 kg of hydrogen was burned. Prior to vessel failure, the hydrogen concentration outside the LOCA node was sustained at 6%. In the node which contained the break peaks as high as 9% were observed, while the sustained concentration was closer to 7%. Prior to vessel failure, the IRWST behaved similar to the bulk of containment. At vessel

DRAFT

failure there were no hydrogen concentration spikes. The hydrogen in the bulk of containment then mixed to a uniform 5% level; however the concentration in the IRWST remained at 6%.

3. In the case with igniters and containment sprays available, a total of 600 kg of hydrogen was burned. The sprays were started shortly after the LOCA occurred; water from the IRWST was sprayed into the dome at a rate of 200 kg/sec. The hydrogen concentration behavior was similar to the non-spray case, with two exceptions. In the node containing the LOCA the peaks never exceeded 7% and the sustained value was 6%; and the IRWST concentration followed the node with the break rather than the bulk of containment. This is attributed to the large sustained inter-node flows driven by the sprays.

7.3.3 Station Blackout with SDS Activation

In these cases the primary system behaved as it would during an extended SBO. Steam and hydrogen were released continuously to the IRWST via an open pressurizer relief valve. The valve was opened at the time of the first relief valve actuation, i.e. well before the core became uncovered. At vessel failure, a small quantity of remaining hydrogen in the RCS was released to the reactor cavity. The total hydrogen released was equivalent to reacting 100% of the active fuel cladding.

1. In the case without igniters available, there were no hydrogen burns. Prior to vessel failure, the hydrogen concentration in the bulk of containment built up to about 11%. In the nodes directly above the IRWST, the hydrogen concentrations were as high as 12%. And in the IRWST, peaks of 30% were observed. After vessel failure, containment nodes mixed to obtain a final hydrogen concentration of between 8% and 9%. The cavity, however, maintained a concentration of 5%.
2. In the case with igniters available, a total of 650 kg of hydrogen was burned. Prior to vessel failure, the hydrogen concentration outside the IRWST built up to about 5%. In the IRWST, peak values of just above 10% were observed, with an average value of about 7%. After vessel failure, containment hydrogen levels dropped to around 4% everywhere in containment.

Combustion at the igniters served to limit the hydrogen concentrations in the IRWST. In the MAAP model, this requires that steam concentrations be below about 55 percent and oxygen concentrations be above 5 percent. Both of these requirements can potentially limit igniter effectiveness. Of course, if the IRWST does become inerted, there is no threat posed by hydrogen build-up, assuming that the igniters remain operational when the atmosphere becomes deinerted later.

Hand calculations indicate that, for all practical purposes, the lower set of IRWST spargers would be utilized just as effectively as

DRAFT

the upper set. Since the lower set of spargers are near the bottom of the tank, it was assumed in these analyses that the bulk of the IRWST would heat uniformly. This would also be promoted if the containment sprays were in operation, since they draw from near the bottom of the tank, or if IRWST cooling is operating.

For this reason, in these calculations oxygen concentrations limited the rate of burning in the IRWST. It was found that oxygen could be continuously supplied to the IRWST by natural convection at a rate sufficient to limit hydrogen concentrations to 10 percent or less. This occurred by one of two mechanisms. If the convective heating of the containment by the steam generators in node 15 (see Figure 7.1-1) was small, hydrogen build-up in node 19 induced circulation out of the IRWST on junction 29 and oxygen was brought in on junction 28. If convective heating was made much larger, as it was in a sensitivity calculation, a large pressure difference developed between nodes 13 and 15 in such a way that the flows to the IRWST became reversed: oxygen was brought in on junction 29 and the burned mixture was swept out on junction 28. In either case, enough oxygen was brought in to limit peak hydrogen concentrations to values of about 10 percent.

3. In the case with both igniters and sprays available, a total of 625 kg hydrogen was burned. As soon as the sprays were started, the containment began to mix vigorously, and a hydrogen concentration of 4% - 5% was rapidly achieved. IRWST hydrogen concentration was sustained at 5%. After vessel failure, the entire containment mixed to a constant 4% in all nodes.

7.4 CONCLUSIONS

If igniters are provided in the containment, hydrogen concentrations outside the IRWST are less than about 10 percent at all times. As expected, hydrogen concentrations are lower than this away from the control volumes containing the IRWST vents and the primary system break, if any. If sprays are in operation, hydrogen concentrations are limited to about 8 percent; this is attributed to the increase in effectiveness of the igniters at low steam concentrations and the more effective inter-node mixing promoted by the operation of the sprays and local combustion.

Igniter effectiveness in the IRWST is sensitive to both steam and oxygen concentrations. Both are considered somewhat uncertain, but the uncertainties act in a direction that would make the mixture non-flammable so as to not present a threat. In these calculations, combustion in the IRWST was limited by oxygen availability. Natural convection of oxygen to the IRWST was induced by the competing effects of hydrogen injection to the IRWST and convective heating of the containment atmosphere above

DRAFT

the IRWST. The calculated flow rates were sufficiently high to maintain combustion at a level that would limit hydrogen concentrations to below about 10 percent.

It is concluded that the System 80+ containment design can adequately deal with even very severe hydrogen source terms without creating conditions that would threaten its integrity.

Table 7.1-1
System 80+ Hydrogen Concentration Cases
Global Containment Values

Case	Maximum Dome Pressure Pa	Maximum Dome Temperature °K	Hydrogen Burned kg	Time to Burn 270* kg sec(from T ₀) sec(from first H ₂ burn)
SLOCA, No Igniters	2.75x10 ⁵	385	0	N/A
SLOCA, Igniters	2.80x10 ⁵	400	500	1.1x10 ⁴ 5.0x10 ³
SLOCA, Igniters, Spray	2.25x10 ⁵	375	600	1.0x10 ⁴ 4.5x10 ³
SBO+PORV, No Igniters	2.30x10 ⁵	375	0	N/A
SBO+PORV, Igniters	2.25x10 ⁵	375	680	9.8x10 ³ 2.0x10 ³
SBO+PORV, Igniters, Spray	2.40x10 ⁵	380	600	1.2x10 ⁴ 3.5x10 ³

* hydrogen mass to be burned to maintain containment global hydrogen concentration below 8%.

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Table 7.1-2
System 80+ Hydrogen Concentration Cases
Peak Hydrogen Concentration

Peak H2 Fraction Case	Cavity	Dome	Lower Compt #2	Steam Generator (low)	IRWST #2	IRWST #3
SLOCA, No Igniters	0.09	0.10	0.10	0.12	0.10	0.10
SLOCA, Igniters	0.06	0.06	0.06	0.09* 0.07**	0.06	0.06
SLOCA, Igniters, Spray	0.06	0.05	0.06	0.06	0.05	0.06
SBO+PORV, No Igniters	0.11	0.11	0.12	0.14	0.30* 0.22**	0.30* 0.22**
SBO+PORV, Igniters	0.05	0.05	0.05	0.05	0.06	0.10-0.11
SBO+PORV, Igniters, Spray	0.05	0.05	0.05	0.05	0.05	0.10* 0.08**

* Spikes
** Average Value

DRAFT

8.0 ASSESSMENT OF THE SYSTEM 80+ HYDROGEN DETONATION ISSUES

This section addresses ancillary concerns associated with the design and operation of the System 80+ hydrogen igniter system.

8.1 LIKELIHOOD OF DETONATION AND CONTAINMENT SURVIVABILITY

As discussed in Section 3.0, even without the availability of a hydrogen igniter system, the accumulation of high concentrations of hydrogen in the System 80+ containment is unlikely due to its large free volume. Furthermore, the basic geometric features of System 80+, are at worst "neutral" to the onset of a detonation via the DDT process and more likely are not conducive to DDT. Based on the simplified Sherman and Berman Ranking Scheme a DDT condition is unlikely. This conclusion is particular true for situations with high steam availability in the containment atmosphere. Consequently, C-E does not believe that detonation within the System 80+ containment is credible. Given that as background, the remainder of this section addresses several "what if" issues associated with the ability of the major containment structures to survive local detonation loadings.

8.1.1 Containment Detonation

During a severe accident, the RCS will release hydrogen at a relatively low point in the containment. All releases to the IRWST and late hydrogen releases via the reactor cavity will enter the bulk containment at or slightly above the 91-9 elevation. For direct containment releases via the RCS hydrogen will typically enter the containment below the 115 foot elevation. These elevations are sufficiently low so as to promote a well mixed containment atmosphere throughout the event. Consequently, concentration gradients are only expected in the vicinity of the source.

Sources of hydrogen are typically limited to:

1. RCS piping including pressurizer surgeline and,
2. IRWST vents

These sources are located within the crane wall. Consequently, the potential for locally detonable mixtures will be in the vicinity of the lower crane wall, pressurizer compartment and the lower portion of the steam generator compartments. An assessment of local detonation loadings for these structures have been performed using the approximate TNT equivalent methodology defined in References 32 and 33. In this analysis, the potential energy release associated with the detonation of a cloud of hydrogen is related to an equivalent TNT point charge and the TNT detonation characteristics are scaled for consideration of the properties of the propagating medium (compared to dry air) and distance of the structure in question from the point source. In this evaluation the

DRAFT

hydrogen gas cloud is assumed to be 50 ft in diameter. The loadings were evaluated 25 feet from the point source. Local hydrogen concentrations from 10 to 15 volume per-cent were considered. Estimated peak pressures, pulse durations and integrated impulse are presented in Table 8-1.

The net impulse loading associated with localized detonations while sufficient to cause damage the walls of the internal structures, the massive supports residing within the IRWST are expected to continue to perform their function. Since all potential detonations are anticipated within the crane wall, no resultant threat to containment is expected since the generated shock loadings will not directly impinge upon the containment shell. Furthermore, the lower mode response frequencies of the containment shell are more than an order of magnitude lower than that associated with the impulse. Thus, dynamic damping of any imposed loading is expected.

DRAFT

TABLE 8.1-1: LOCAL HYDROGEN DETONATION PRESSURES: TNT EQUIVALENT			
HYDROGEN VOLUME PERCENT	PEAK PRESSURE (PSIA)	PULSE DURATION (MSEC)	IMPULSE (PSI-SEC)
10	230	6.5	0.478
13	280	7	0.578
15	310	7.7	0.669

DRAFT

8.2 COMMENTS ON THE POTENTIAL FOR A CONDENSATION INDUCED
DETONATION

One serious concern with regard to the operation of an igniter system, is the potential system response following a rapid spray induced condensation of steam late in a severe accident scenario. This situation may arise as a consequence of spray recovery in a sequence where the steam inerting prevented proper operation of the igniter system. Thus, high hydrogen: air concentrations, will develop along with low steam concentration. Ignition of this mixture is virtually assured via the HMS.

Experimental evidence to date does not justify this concern. Hydrogen combustion experiments performed in the presence of a condensing environment indicate that igniters will initiate combustion in the form of a deflagration as the mixture passes through the mixture flammability limit. While these experiments are not prototypical of System 80+, it is believed to be generally applicable to reactors provided the condensation process is over a several minute (as opposed to several second) time interval. Intervals of several minutes are nearly quasi-steady from the viewpoint of combustion initiation. Analyses performed for System 80+ confirm that for the "worst case" limiting assumptions of a localized condensation from a minimum inserted steam state, to a potentially detonable state indicate the system will take over 3.5 minutes prior to becoming minimally detonable. Consequently, deflagrations have sufficient time to precede detonations.

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ATTACHMENT 2

CESSAR-DC Ch.16 TECH SPECS

NOTES ON CHANGES TO STANDARD TECH SPECS TO ADDRESS SYSTEM 80+

- ① LCO and BASES FOR 3.1.1 and 3.1.2 are combined because SAFETY ANALYSES supporting plant operation do not distinguish between $T_{avg} > 200^{\circ}\text{F}$ and $T_{avg} \leq 200^{\circ}\text{F}$ in terms of shutdown margin.
- ② Notes b1 and b2 are added to LCO 3.1.1 to ADDRESS more stringent core reactivity requirements arising from higher CEA Bank worths in SYSTEM 80+ DESIGN.
- ③ The following sections are revised to address the use, in the SYSTEM 80+ DESIGN, of FULL-LENGTH PART-STRENGTH CEAs, as opposed to Part-length CEAs (as discussed in the STANDARD TECH. SPECS):
 LCO^{SECTION OF} 3.1.5, 3.1.8, 3.1.9, 3.1.10
 BASES SECTIONS OF 3.1.1, 3.1.5, 3.1.7, 3.1.8, 3.1.9, 3.1.10, 3.2.1, 3.2.2, 3.2.3, 3.2.4, 3.2.5
- ④ SYSTEM 80+ -SPECIFIC figure (3.1.5-1), "REQUIRED POWER REDUCTION after CEA Deviation" added to LCO 3.1.5
- ⑤ LCO and BASES SECTION 3.1.12, "Boron Dilution Alarm" added to ensure proper alarm functioning of this system.
- ⑥ LCO and BASES SECTION 3.1.11, "Special Test Exception - CEDMS TESTING" added to allow demonstration of the operability of the control element drive mechanism system.

- ⑦ BASES FOR 3.1.1 modified to REFLECT THAT SAFETY ANALYSES FOR SYSTEM 80+ MSLB PREDICT THAT A POST-TRIP RETURN TO POWER WILL NOT OCCUR.
- ⑧ BASES FOR 3.1.1 modified to REFLECT THAT, FOR SYSTEM 80+, THE UNCONTROLLED CEA WITHDRAWAL TRANSIENT WILL NOT BE TERMINATED BY A HIGH PRESSURIZED PRESSURE TRIP AND MAY BE TERMINATED BY A LOW DNBTR TRIP OR A HIGH LOCAL POWER DENSITY TRIP, IN ADDITION TO A HIGH POWER LEVEL TRIP.
- ⑨ BASES FOR 3.1.4 modified to address the capability of the SYSTEM 80+ design to accommodate distributed burnable poisons that are NOT lumped or fixed IN the CORE (eg. $Gd_2O_3-UO_2$ or $Er_2O_3-UO_2$ fuel rods)
- ⑩ BASES FOR 3.1.4 modified to CORRECT A TYPOGRAPHICAL ERROR IN the first paragraph of the APPLICABILITY subsection.
- ⑪ BASES FOR 3.1.5 modified to reflect the capability of the SYSTEM 80+ DESIGN to USE Full-STRENGTH RODS for daily load follow as well as for axial shape control.
- ⑫ BASES FOR 3.1.5 modified to INCLUDE DISCUSSION OF THE (ACCEPTABLE) CONSEQUENCES OF A full-strength CEA subgroup drop and a part-strength CEA drop, as demonstrated by the SYSTEM 80+ SAFETY ANALYSES

- ⑬ Bases of 3.1.6 modified to remove first paragraph of SURVEILLANCE REQUIREMENT 3.1.6.1 discussion, SINCE VERIFICATION OF SHUTDOWN SEA INSERTION LIMITS WITHIN 15 MINUTES PRIOR TO AN APPROACH TO CRITICALITY IS NOT PART OF THE LCD FOR THIS TECHNICAL SPECIFICATION.
- ⑭ Bases of 3.1.10 is modified to include an explanation of the ~~aka~~ selection of a power plateau of <85% RTP for performing Physics Tests.
- ⑮ SN 3.2.3.3 of 3.2.3 is modified from a frequency of 31 EFPD to 31 days in accordance with the description in the Bases for 3.2.3.
- ⑯ Bases for 3.2.1 in ACTION B.1 description is modified to correct CPE monitoring of LHR from every 15 minutes to every 2 hours to be consistent with the LCD for 3.2.1.
- ⑰ Same as item ⑯ for 3.2.4

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM) ~~$T_{avg} > 200^{\circ}\text{F}$ (Digital)~~

LCO 3.1.1 a. SDM shall be \geq ^{6.5}~~5.0~~ % $\Delta k/k$; and either

b1. With reactor trip circuit breakers (RTCBs) closed: the estimated position shall be within the limits of LCOs 3.1.6 and 3.1.7; or

APPLICABILITY: MODES 3, 4, and 5

b2. With RTCBs open: K_{eff} shall be < 0.99 .

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM is \geq ^{6.5} 5.0 % $\Delta k/k$.	24 hours

Combining 3.1.1 & 3.1.2

CESSAN-DC says SEE STE 3.1.8 & 3.1.10 for applicability notes

$$\text{SDM} - T_{\text{avg}} \leq 200^{\circ}\text{F (Digital)}$$

3.1.2

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 SHUTDOWN MARGIN (SDM) - $T_{\text{avg}} \leq 200^{\circ}\text{F (Digital)}$ [DELETED]

LCO 3.1.2 SDM shall be $\geq [2.0]\% \Delta k/k$.

APPLICABILITY: MODE 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.2.1 Verify SDM is $\geq [2.0]\% \Delta k/k$.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Reactivity Balance (Digital)

LCO 3.1.3 The core reactivity balance shall be within $\pm 1\% \Delta k/k$ of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core reactivity balance not within limit.	A.1 Re-evaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation.	72 hours
	AND A.2 Establish appropriate operating restrictions and SRs.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.3.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. 2. This Surveillance is not required to be performed prior to entry into MODE 2. <p>-----</p> <p>Verify overall core reactivity balance is within $\pm 1.0\% \Delta k/k$ of predicted values.</p>	<p>Prior to entering MODE 1 after fuel loading</p> <p>AND</p> <p>-----NOTE----- Only required after 60 EFPD -----</p> <p>31 EFPD</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Moderator Temperature Coefficient (MTC) (~~Digital~~)

LCO 3.1.4 The MTC shall be maintained within the limits specified in the COLR, and a maximum positive limit as specified below:

- a. $[0.5 \text{ E-}4] \Delta k/k/^\circ\text{F}$ when THERMAL POWER is $\leq 70\%$ RTP; and
b. $[0.0] \Delta k/k/^\circ\text{F}$ when THERMAL POWER is $> 70\%$ RTP.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within limits.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 -----NOTE----- This Surveillance is not required to be performed prior to entry into MODE 2. ----- Verify MTC within the upper limit specified in the COLR.	Prior to entering MODE 1 after each fuel loading

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.1.4.2 -----NOTES-----</p> <ol style="list-style-type: none">1. This Surveillance is not required to be performed prior to entry into MODE 1 or 2.2. If the MTC is more negative than the COLR limit when extrapolated to the end of cycle, SR 3.1.4.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. <p>-----</p> <p>Verify MTC is within the lower limit specified in the COLR.</p>	<p>Each fuel cycle within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup</p> <p><u>AND</u></p> <p>Each fuel cycle within 7 EFPD of reaching $\frac{2}{3}$ of expected core burnup</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Element Assembly (CEA) Alignment (Digital)

LCO 3.1.5 All full length CEAs shall be OPERABLE, and all full and part length CEAs shall be aligned to within [7 inches] (indicated position) of their respective groups.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more regulating CEAs trippable and misaligned from its group by > [7 inches] and ≤ [19 inches]. <u>OR</u> One regulating CEA trippable and misaligned from its group by > [19 inches].	A.1 Reduce THERMAL POWER in accordance with Figure 3.1.5-1.	1 hour
	<u>AND</u>	
	A.2.1 Verify SDM is $\geq [5.0] \Delta k/k$. 6.5	1 hour
	<u>OR</u>	
	A.2.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.3.1 Restore the misaligned CEA(s) to within [7 inches] (indicated position) of its group.	2 hours
	<u>OR</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3.2 Align the remainder of the CEAs in the group to within [7 inches] (indicated position) of the misaligned CEA(s) while maintaining the insertion limit of LCO 3.1.7. "Regulating Control Element Assembly (CEA) Insertion Limits."	2 hours
B. One or more shutdown CEAs trippable and misaligned from its group by > [7 inches] and \leq [19 inches]. <u>OR</u> One shutdown CEA trippable and misaligned from its group by > [19 inches].	B.1 Reduce THERMAL POWER in accordance with Figure 3.1.5-1. <u>AND</u> B.2.1 Verify SDM is \geq [5.0] % $\Delta k/k$. 6.5 <u>OR</u> B.2.2 Initiate boration to restore SDM to within limit. <u>AND</u> B.3 Restore the misaligned CEA(s) to within [7 inches] (indicated position) of its group.	1 hour 1 hour 1 hour 2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. One or more part length ^{Strength} CEAs misaligned from its group by > [7 inches] and ≤ [19 inches].</p> <p><u>OR</u></p> <p>One part length ^{Strength} CEA misaligned from its group by > [19 inches].</p>	<p>C.1 Reduce THERMAL POWER in accordance with Figure 3.1.5-1.</p> <p><u>AND</u></p> <p>C.2.1 Restore the misaligned CEA(s) to within [7 inches] (indicated position) of its group.</p> <p><u>OR</u></p> <p>C.2.2 Align the remainder of the CEAs in the group to within [7 inches] (indicated position) of the misaligned CEA(s).</p>	<p>1 hour</p> <p>2 hours</p> <p>2 hours</p>
<p>D. Required Action and associated Completion Time of Condition A, Condition B, or Condition C not met.</p> <p><u>OR</u></p> <p>Strength ^{Strength} One or more full length CEAs untrippable.</p> <p><u>OR</u></p> <p>Two or more CEAs misaligned by > [19 inches].</p>	<p>D.1 Be in MODE 3.</p>	<p>6 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5.1 Verify the indicated position of each full and part length CEA is within [7 inches] of all other CEAs in its group.	12 hours
SR 3.1.5.2 Verify that, for each CEA, its OPERABLE CEA position indicator channels indicate within [5 inches] of each other.	12 hours
SR 3.1.5.3 Verify full length CEA freedom of movement (trippability) by moving each individual full length CEA that is not fully inserted in the core at least [5 inches].	92 days
SR 3.1.5.4 Perform a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel.	[18 months]
SR 3.1.5.5 Verify each full length CEA drop time \leq [3.5] seconds and the arithmetic average of all full length CEA drop times \leq [3.2] seconds.	Prior to reactor criticality, after each removal of the reactor head

-----NOTE-----

When core power is reduced to 60% RTP per this limit curve, further reduction is not required by this Specification.

*add figure from CESSAR
HERE (Pg. 16.4-12)
- del next page -*

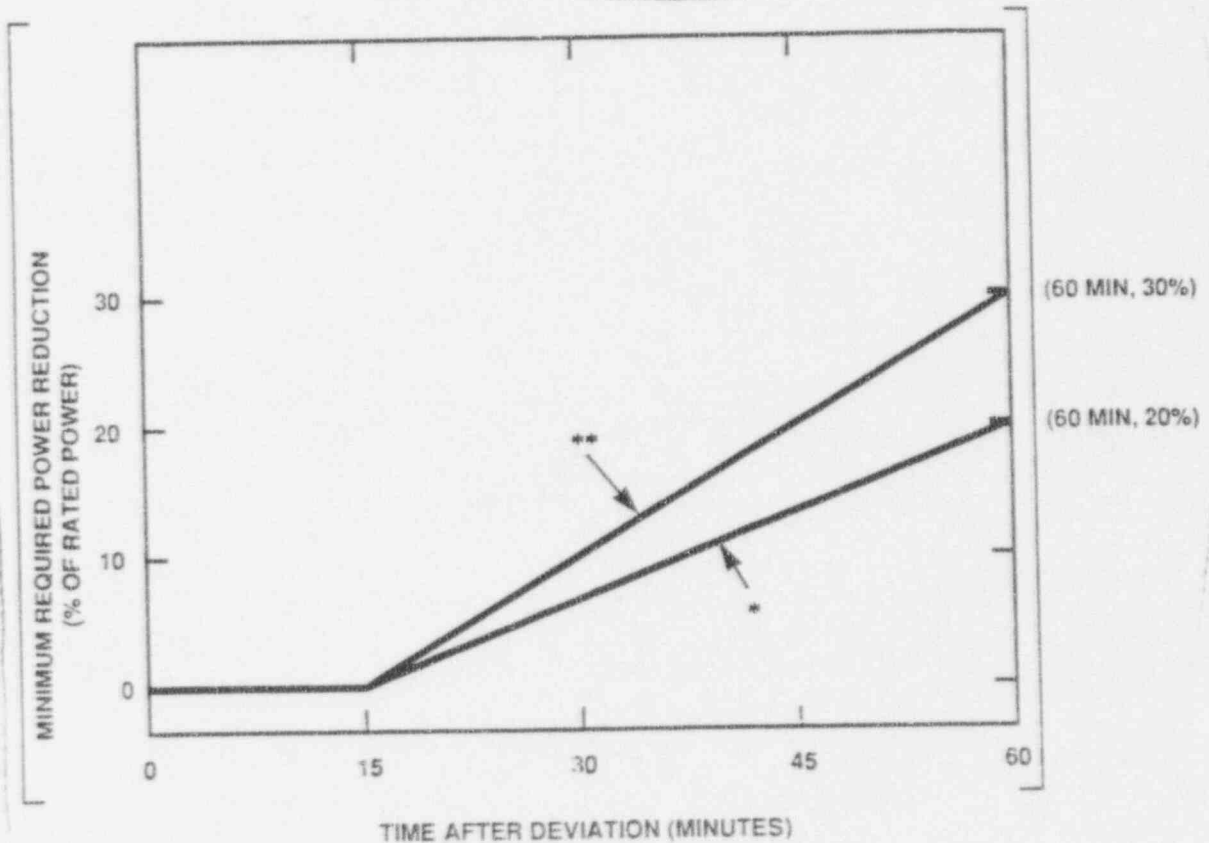
NOT TO BE USED FOR OPERATION.
FOR ILLUSTRATION PURPOSES ONLY.

Figure 3.1.5-1 (page 1 of 1)
Required Power Reduction After CEA Deviation

CEA Alignment
3.1.5

NOTE

When core power is reduced to 60% RTP per this limit curve, further reduction is not required by this specification.



- * CEA Misalignment for either Bank [3], Bank [P1], or Bank [P2].
- ** CEA Misalignment for any Bank not mentioned in the previous note.

REQUIRED POWER REDUCTION AFTER CEA DEVIATION

FIGURE 3.1.5-1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Shutdown Control Element Assembly (CEA) Insertion Limits (Digital)

LCO 3.1.6 All shutdown CEAs shall be withdrawn to \geq [145] inches.APPLICABILITY: MODE 1.
MODE 2 with any regulating CEA not fully inserted.-----NOTE-----
This LCO is not applicable while performing SR 3.1.5.3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown CEAs not within limit.	A.1.1 Verify SDM \geq [50] % $\Delta k/k$. 65	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore shutdown CEA(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.6.1 Verify each shutdown CEA is withdrawn \geq [145] inches.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Regulating CEA Insertion Limits (Digital)

- LCO 3.1.7 The power dependent insertion limit (PDIL) alarm circuit shall be OPERABLE, and
- With the Core Operating Limit Supervisory System (COLSS) in service, the regulating CEA groups shall be limited to the withdrawal sequence, insertion limits, and associated time restraints specified in the COLR.
 - With COLSS out of service, the regulating CEA groups shall be limited to the short term steady state insertion limit and associated time restraints specified in the COLR.

APPLICABILITY: MODES 1 and 2.

-----NOTE-----
This LCO is not applicable while conducting SR 3.1.5.3 [or during reactor power cutback operation].

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Regulating CEA groups inserted beyond the transient insertion limit with COLSS in service.	A.1.1 Verify SDM $\geq [5.0] \Delta k/k$ 6.5	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2.1 Restore regulating CEA groups to within limits.	2 hours
	OR	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Reduce THERMAL POWER to less than or equal to the fraction of RTP allowed by the CEA group position and insertion limits specified in the COLR.	2 hours
B. Regulating CEA groups inserted between the long term steady state insertion limit and the transient insertion limit for > 4 hours per 24 hour interval with COLSS in service.	B.1 Verify short term steady state insertion limits are not exceeded.	15 minutes
	<u>OR</u> B.2 Restrict increases in THERMAL POWER to $\leq 5\%$ RTP per hour.	15 minutes
C. Regulating CEA groups inserted between the long term steady state insertion limit and the transient insertion limit for intervals > 5 effective full power days (EFPD) per 30 EFPD interval or > 14 EFPD per 365 EFPD interval with COLSS in service.	C.1 Restore regulating CEA groups to within limits.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Regulating CEA groups inserted beyond the short term steady state insertion limit with COLSS out of service.	D.1.1 Verify SDM $\geq [20] \Delta k/k$ 6.5	1 hour
	<u>OR</u>	
	D.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	D.2.1 Restore regulating CEA groups to within limits.	2 hours
	<u>OR</u>	
	D.2.2 Reduce THERMAL POWER to less than or equal to the fraction of RTP allowed by CEA group position and short term steady state insertion limit specified in the COLR.	2 hours
E. PDIL alarm circuit inoperable.	E.1 Perform SR 3.1.7.1.	1 hour
		<u>AND</u> Once per 4 hours thereafter
F. Required Actions and associated Completion Times not met.	F.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	<p>-----NOTE----- This Surveillance is not required to be performed prior to entry into MODE 2. -----</p> <p>Verify each regulating CEA group position is within its insertion limits.</p>	12 hours
SR 3.1.7.2	Verify the accumulated times during which the regulating CEA groups are inserted beyond the steady state insertion limits but within the transient insertion limits.	24 hours
SR 3.1.7.3	Verify PDIL alarm circuit is OPERABLE.	31 days

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Part Length Control Element Assembly (CEA) Insertion Limits (Optional) (Digital)

LCO 3.1.8 The part length CEA groups shall be limited to the insertion limits specified in the COLR.

APPLICABILITY: MODE 1 > 20% RTP.

-----NOTE-----
This LCO not applicable while exercising part length CEAs.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Part length CEA groups inserted beyond the transient insertion limit.	A.1 Restore part length CEA groups to within the limit.	2 hours
	OR A.2 Reduce THERMAL POWER to less than or equal to that fraction of RTP specified in the COLR.	2 hours
B. Part length CEA groups inserted between the long term steady state insertion limit and the transient insertion limit for intervals ≥ 7 effective full power days (EFPD) per 30 EFPD or ≥ 14 EFPD per 365 EFPD interval.	B.1 Restore part length CEA groups to within the long term steady state insertion limit.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Reduce THERMAL POWER to $\leq 20\%$ RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.8.1 Verify part length CEA group position.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 Special Test Exception (STE) - SHUTDOWN MARGIN (SDM) (Digital)

LCO 3.1.9

The SDM requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM) - $T_{avg} > 200^{\circ}F$ " and the regulating control element assembly (CEA) insertion limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits," may be suspended for measurement of CEA worth and SDM, provided shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) is available for trip insertion.

APPLICABILITY: MODES 2 and 3 during PHYSICS TESTS.

-----NOTE-----
Operation in MODE 3 shall be limited to 6 consecutive hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any full <u>92</u> length CEA not fully inserted and less than the required shutdown reactivity available for trip insertion. OR All full <u>92</u> length CEAs inserted and the reactor subcritical by less than the above required shutdown reactivity equivalent.	A.1 Initiate boration to restore required shutdown reactivity.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.9.1 Verify that the position of each CEA not fully inserted is within the acceptance criteria for available negative reactivity addition.	2 hours
SR 3.1.9.2 Verify each full <u>length</u> CEA not fully inserted is capable of full insertion when tripped from at least the 50% withdrawn position.	Within [7 days] prior to reducing SDM to less than the limits of LCO 3.1.1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.10 Special Test Exceptions (STE) - MODES 1 and 2 (Digital)

LCO 3.1.10 During performance of PHYSICS TESTS, the requirements of:

- LCO 3.1.4, "Moderator Temperature Coefficient (MTC)";
- LCO 3.1.5, "Control Element Assembly (CEA) Alignment";
- LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits";
- LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits";
- LCO 3.1.8, "Partial Length CEA Insertion Limits";
- LCO 3.2.2, "Planar Radial Peaking Factors"; and
- LCO 3.2.3, "AZIMUTHAL POWER TILT (Tq)"

may be suspended, provided:

- a. THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP; and
- b. SDM is \geq ~~2.0~~ % $\Delta k/k$.

6.5

APPLICABILITY: MODES 1 and 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Test power plateau exceeded.	A.1 Reduce THERMAL POWER to less than or equal to the test power plateau.	15 minutes

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. SDM is not within limit.	B.1 Initiate boration to restore SDM to within limit.	15 minutes
	<u>AND</u> B.2 Suspend PHYSICS TESTS.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Suspend PHYSICS TESTS.	6 hours
	<u>AND</u> C.2 Be in MODE 3.	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.10.1 Verify THERMAL POWER equal to or less than the test power plateau.	1 hour
SR 3.1.10.2 Verify SDM is \geq 5.0 _{6.5} % $\Delta k/k$.	24 hours

ADD

LCO'S

3.1.11 = CESSAR-DC
LCO 3.1.10
"LEDMS TESTING"

AND

3.1.12 = CESSAR-DC
LCO 3.1.11
"BORON DILUTION ALARMS"

as shown on attached pages

16.4.10

3.1.10 SPECIAL TEST EXCEPTIONS - CEDMS TESTING

STE-CEDMS Testing

3.1.10

3.1 REACTIVITY CONTROL SYSTEMS

3.1.10 Special Test Exception-CEDMS Testing

LCO 3.1.10 The SHUTDOWN MARGIN requirement of Specification 3.1.1 may be suspended for pre-startup tests to demonstrate the OPERABILITY of the control element drive mechanism system (CEDMS) provided:

- No more than one CEA is withdrawn at any time.
- No CEA is withdrawn more than [seven] inches.
- With RTCBs open, K_{N-1} shall be less than [0.99] prior to the start of testing.
- All other operations involving positive reactivity changes are suspended during the testing.

APPLICABILITY: MODES 4 and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any of the above requirements not met.	A.1 Suspend testing and initiate boration to restore SDM to within the limit of LCO 3.1.1.	Immediately

SYSTEM 80+

3.1-25

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.10.1 Determine SDM	Once per 24 hours

* Consider the following factors:

1. RCS boron concentration
2. CEA position
3. RCS average temperature
4. Fuel burnup based on gross thermal energy generation
5. Xenon concentration
6. Samarium concentration

16.4.11

¹² 3.1.11 BORON DILUTION ALARMS

Boron Dilution Alarm

3.1.11

¹²

3.1 REACTIVITY CONTROL SYSTEMS

¹² 3.1.11 Boron Dilution Alarms

LCO ¹² 3.1.11 Both startup channel high neutron flux alarms shall be operable.

APPLICABILITY: MODES 3*, 4, 5, and 6.

* Within 1 hour after the neutron flux is within the startup range following a reactor shutdown.

*Good due to in function
in the startup
of the system
to be checked
for the system
to be checked
for the system*

SYSTEM 80+

3.1-27

Boron Dilution Alarm

3.1.12

12

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One startup channel high neutron flux alarm inoperable.	A.1 Initiate action to restore the inoperable channel to OPERABLE status.	Immediately
	AND	
	A.2 Determine the RCS boron concentration when entering Mode 3,4,5 or 6 or at the time the alarm is determined inoperable.	Immediately
	AND	Once per required frequency identified in Tables 3.1.11-1 through 3.1.11-5.
B. Both startup channel high neutron flux alarms inoperable.	B.1 Initiate action to restore a single channel to OPERABLE status.	Immediately
	AND	
	B.2 Determine the RCS boron concentration when entering Mode 3, 4, 5 or 6, or at the time both alarms are determined inoperable.	Immediately
	AND	Once per required frequency identified in Tables 3.1.11-1 through 3.1.11-5.
	B.3 Suspend all operations involving CORE ALTERATIONS or positive reactivity changes.	Immediately

SYSTEM 80+

3.1-28

Boron Dilution Alarm

3.1.14

12

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
3.1.14.1	Perform a CHANNEL CHECK.	Once per 12 hours**
3.1.14.2	Perform a CHANNEL CALIBRATION	Every 31 days of cumulative operation during shutdown.

** When initially setting setpoints at the following times:

- a) One hour after a reactor trip
- b) After a controlled reactor shutdown: Within 1 hour after the neutron flux is within the startup range in Mode 3.

SYSTEM 80+

3.1-29

12
TABLE 3.1.12-1

**REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON
DILUTION DETECTION AS A FUNCTION OF OPERATING
CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $K_{eff} > 0.98$**

OPERATIONAL MODE	Number of Operating Charging Pumps	
	0	1
3	[12 hours]	[1 hour]
4 not on SCS	[12 hours]	[1 hour]
5 not on SCS	[8 hours]	[1 hour]
4 & 5 on SCS	[ONA]	[ONA]

12
TABLE 3.1.12-2

**REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION
DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT
OPERATIONAL MODES FOR $0.98 \geq K_{eff} > 0.97$**

OPERATIONAL MODE	Number of Operating Charging Pumps	
	0	1
3	[12 hours]	[2.0 hours]
4 not on SCS	[12 hours]	[2.5 hours]
5 not on SCS	[8 hours]	[2.5 hours]
4 & 5 on SCS	[8 hours]	[0.5 hours]

Notes: SCS = Shutdown Cooling System
ONA = Operation Not Allowed

TABLE 3.1.41-3

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION
DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS
AND PLANT OPERATIONAL MODES FOR $0.97 \geq K_{eff} > 0.96$

OPERATIONAL MODE	Number of Operating Charging Pumps	
	0	1
3	[12 hours]	[3.5 hours]
4 not on SCS	[12 hours]	[3.5 hours]
5 not on SCS	[8 hours]	[3.5 hours]
4 & 5 on SCS	[8 hours]	[1 hour]

TABLE 3.1.41-4

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION
DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS
AND PLANT OPERATIONAL MODES FOR $0.96 \geq K_{eff} > 0.95$

OPERATIONAL MODE	Number of Operating Charging Pumps	
	0	1
3	[12 hours]	[5 hours]
4 not on SCS	[12 hours]	[5 hours]
5 not on SCS	[8 hours]	[5 hours]
4 & 5 on SCS	[8 hours]	[2 hours]

Notes: SCS = Shutdown Cooling System

Boron Dilution Alarm

3.1.12

12.

¹²
TABLE 3.1.12-5

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION
DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS
AND PLANT OPERATIONAL MODES FOR $K_{eff} \leq 0.95$

OPERATIONAL MODE	Number of Operating Charging Pumps	
	0	1
3	[12 hours]	[6 hours]
4 not on SCS	[12 hours]	[6 hours]
5 not on SCS	[8 hours]	[6 hours]
4 & 5 on SCS	[8 hours]	[2 hours]
6	[24 hours]	[8 hours]

Notes: ~~SCS~~ = Shutdown Cooling System

STU W
T, Th, AFT
F MON

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM) $\left(\begin{array}{c} \text{BVG} \\ \text{BVG} \end{array} \right) > 200^{\circ}\text{F (Digital)}$

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality which would be obtained immediately following the insertion of all full-length control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the CEAs, together with the boration system, provide the SDM during power operation and are capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the CEA of highest reactivity worth remains fully withdrawn. *K_{eff} is defined as the reactivity of the core with all CEAs inserted assuming that the CEA of highest reactivity worth remains fully withdrawn.*

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown CEAs fully withdrawn and the regulating CEAs within the limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

* ADD SENTENCE AT
END OF SET 9 FROM
B3.1.2 SAFETY
ANAL. SECTION, Pg.
B3.1-8

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AODs, with the assumption of the highest worth CEA stuck out following a reactor trip.*

The acceptance criteria for the SDM are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

- The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;
- The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limit, AODs, and ≤ 280 cal/gm energy deposition for the CEA ejection accident); *and* *for*
- The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements are based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power ~~will not~~ *may* occur; however, no fuel damage occurs as a result of the post trip return to

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

^{W:11}
~~power~~, and THERMAL POWER ~~does~~ not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

- Inadvertent boron dilution;
- An uncontrolled CEA withdrawal from a subcritical or low power condition;
- Startup of an inactive reactor coolant pump (RCP); and
- CEA ejection.

Each of these is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

ADD LAST ^{3 Full}
9/4 FROM B3.1.2
PS. B3.1-8

The withdrawal of CEAs from subcritical or low power conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either a high power level trip ^{Low DNBR trip, a high temp. To be seen in P&R, etc.} or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition which can occur due to an inadvertent RCP start is less than half the minimum required

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

SDM. An idle RCP cannot, therefore, produce a return to power from the hot standby condition.

~~or ejection~~
The withdrawal of CEAs from subcritical or low power conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power.

Replace with L&B
4 FROM B3.1-2 PG.
B3.1-8

The SDM satisfies Criterion 2 of the NRC Policy Statement.

LCO ADD
1ST A
FROM
LCO 5
OF
B3.1.2

~~accident~~ accident initiated in Mode 5
The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criterion," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

Add 2+3 from LCO 3.1.2
P. 63.1-5 - SEE
1876-

SDM is a core physics design condition that can be ensured through CFA positioning (regulating and shutdown CEAs) and through the soluble boron concentration.

APPLICABILITY

In MODES ~~3 and 4~~ ^{3 and 5}, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7. If the insertion limits of LCO 3.1.6 or LCO 3.1.7 are not being complied with, SDM is not automatically violated. The SDM must be calculated by performing a reactivity balance calculation (considering the listed reactivity effects in Bases Section 3.1.1.1). ~~In MODE 5, SDM is addressed by LCO 3.1.2, "Shutdown Margin (SDM) - $T_{avg} < 200^{\circ}\text{F}$."~~ In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

(continued)

Shutdown Margin
B 3.1.1

BASES

LCO

from SRS 3.1.2

The accident analysis has shown that the required SDM is sufficient to avoid unacceptable consequences to the fuel or RCS as a result of the events addressed above. The zero power MSLE establishes this value. Shutdown boron concentration requirements assume the highest worth CEA is stuck in the fully withdrawn position to account for a postulated inoperable or untrippable CEA prior to reactor shutdown.

*from
SRS
3.1.1
b.1
b.2
in SRS*

LCO 3.1.1 b.1 requires that the calculated critical position be within the limits of Technical Specifications 3.1.6 and 3.1.7 when the reactor trip breakers are closed. This ensures that the most adverse subcritical CEA withdrawal event scenario is the inadvertent withdrawal of a regulating CEA bank, i.e. the reactor will not become critical due to the inadvertent withdrawal of a shutdown CEA bank.

LCO 3.1.1 b.1 also ensures that, if the RTCBs are closed, a CEA ejection event postulated to be initiated at these conditions would result in less net positive reactivity insertion than for a case initiated from a critical position. If the RTCBs are open, LCO 3.1.1 b.2 requires that the value of k_{eff} must remain less than 1.0 when the highest worth CEA is excluded from the calculation. This, latter, requirement ensures that the reactor would not reach criticality for a CEA ejection event postulated to be initiated under these conditions. Together, therefore, LCO 3.1.1 b.1 and LCO 3.1.1 b.2 ensure that a CEA ejection event postulated to be initiated in MODE 2 subcritical or MODEs 3, 4, or 5 would have less adverse consequences than the event analyzed for a CEA ejection in MODE 1, which has been shown to have acceptable consequences (Ref. 2).

The LCO has been modified by a Note which states the CEA of highest reactivity worth does not have to be assumed withdrawn when all CEAs are verified inserted by two diverse position indicators. This means the worth of the most reactive CEA does not have to be included in the calculation of SDM thus eliminating unnecessary boration and dilution.

(continued)

BASES (continued)

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank or the borated water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of [] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of [] gpm and [] ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE
REQUIREMENTSSR 3.1.1.1

SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

- RCS boron concentration:
- CEA positions:
- RCS average temperature:

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.1.1.1 (continued)

- d. Fuel burnup based on gross thermal energy generation;
- e. Xenon concentration;
- f. Samarium concentration; and
- g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the ^{that of} generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

- 1. 10 CFR 50, Appendix A, GDC 26.
- 2. FSAR, Section [15.4.2].
- 3. FSAR, Section [15.4.2].
- 4. 10 CFR 100.

$$\text{SDM} - T_{\text{avg}} \leq 200^{\circ}\text{F (Digital)}$$

B 3.1.2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 SHUTDOWN MARGIN (SDM) - $T_{\text{avg}} \leq 200^{\circ}\text{F (Digital)}$

[DELETED]

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, SDM defines the degree of subcriticality which would be obtained immediately following the insertion of all control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the CEAs, together with the boration system, provide the SDM during power operation and are capable of making the core subcritical rapidly enough to prevent exceeding the acceptable fuel damage limits, assuming that the CEA of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown CEAs fully withdrawn and the regulating CEAs within the limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

(continued)

[DELETED]

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs with the assumption of the highest worth CEA stuck out following a reactor trip. Specifically, for MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

ADD TO B 3.1-1
INFC B 3.1-2

The acceptance criteria for the SDM requirements are that the specified acceptable fuel design limits are maintained. This is done by ensuring that:

- a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;
- b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio, fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm energy deposition for the CEA ejection accident); and
- c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

An inadvertent boron dilution is a moderate Frequency incident as defined in Reference 2. The core is initially subcritical with all CEAs inserted. A Chemical and Volume Control System malfunction occurs which causes unborated water to be pumped to the RCS via ~~three~~ ^{one} charging pump.

Rel to B 3.1-1
Pg. B 3.1-3

The reactivity change rate associated with boron concentration changes due to inadvertent dilution is within the capabilities of operator recognition and control.

The high neutron flux alarm on the startup channel instrumentation will alert the operator of the boron dilution with a minimum of 15 minutes remaining before the core becomes critical.

ADD TO B 3.1-1
Pg. B 3.1-4

SDM satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of the accident analysis assumptions.

(continued)

[DELETED]

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

is operating within the bounds of accident analysis assumptions.

LCO

The accident analysis has shown that the required SDM is sufficient to avoid unacceptable consequences to the fuel or RCS as a result of the events addressed above.

ADD to B3.1-1
Pg. B3.1-4

The boron dilution (Ref. 2) accident initiated in MODE 5 is the most limiting analysis which establishes the SDM value of the LCO. For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

SDM is a core physics design condition that can be ensured through CEA positioning (regulating and shutdown CEAs) and through soluble boron concentration.

APPLICABILITY

In MODE 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7. If the insertion limits of LCO 3.1.6 or LCO 3.1.7 are not being complied with, SDM is not automatically violated. The SDM must be calculated by performing a reactivity balance calculation (considering the listed reactivity effects in Bases Section SR 3.1.2.1). In MODES 1, 2, 3, and 4, the SDM requirements are given in LCO 3.1.1, "SHUTDOWN MARGIN (SDM) - $T_{avg} > 200^{\circ}\text{F}$." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that

(continued)

BASES

ACTIONS

A.1 (continued)

boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank or the borated water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate the time core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of $1\% \Delta k/k$ must be recovered and a boration flow rate of [] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by $1\% \Delta k/k$. These boration parameters of [] gpm and [] ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.1

In MODE 5 the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

- a. RCS boron concentration;
- b. CEA positions;
- c. RCS average temperature;
- d. Fuel burnup based on gross thermal energy generation;
- e. Xenon concentration;

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.1 (continued)

- f. Samarium concentration; and
- g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as that of the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration, and it allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

- 1. 10 CFR 50. Appendix A. GDC 26.
 - 2. FSAR, Section [15.2.4].
-

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Reactivity Balance (Digital)

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that, subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accidents (DBA) and transients safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control element assembly (CEA) worth, or operation at Conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ≥ 1.0 (avg) $\geq 200^\circ\text{F}$ ") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the critical boron curve, which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as CEA height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations, and that the calculational models used to generate the safety analysis are adequate.

(continued)

BASES

BACKGROUND
(continued)

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), CEAs, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The critical boron curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE
SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM, and reactivity transients, such as CEA withdrawal accidents or CEA ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted critical boron curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the CEAs in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

The reactivity balance satisfies Criterion 2 of the NRC Policy Statement.

LCU

The reactivity balance limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity balance of $\pm 1\% \Delta k/k$ has been established, based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation, and should therefore be evaluated.

When measured core reactivity is within $1\% \Delta k/k$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached.

(continued)

BASES

LCO
(continued)

These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down, and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, or CEA replacement, or shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized, and power operation may continue. If operational restrictions or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the $1\% \Delta k/k$, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.3.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable including CEA position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.3.1 (continued)

is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by three Notes. The first Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (e.g., QPTR) for prompt indication of an anomaly. A Note, "only required after 60 EFPD," is added to the Frequency column to allow this. Another Note indicates that the performance of SR 3.1.3.1 is not required prior to entering MODE 2. This Note is required to allow a MODE 2 entry to verify core reactivity because Applicability is for MODES 1 and 2.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
 2. FSAR, Section [15].
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Moderator Temperature Coefficient (MTC) (~~Digital~~)BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature. A positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature. The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result. The same characteristic is true when the MTC is positive and coolant temperature decreases occur.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less positive than that allowed by the LCO. The actual value of the MTC is dependent on core characteristics such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional ~~fixed~~ distributed poisons (~~lumped~~ burnable poison assemblies) to yield an MTC at the BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

- a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and
- b. The MTC must be such that inherently stable power operations result during normal operation and during accidents, such as overheating and overcooling events.

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations, to ensure the accident results are bounding (Ref. 3).

Accidents that cause core overheating, either by decreased heat removal or increased power production, must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the control element assembly (CEA) withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to a positive MTC is a CEA withdrawal accident from zero power, also referred to as a startup accident (Ref. 4).

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction combined with the large negative MTC may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power is produced with all CEAs inserted, except the most reactive one, which is assumed withdrawn. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. A middle of cycle (MOC) measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

The MTC satisfies Criterion 2 of the NRC Policy Statement.

LCO

LCO 3.1.4 requires the MTC to be within the specified limits of the COLR to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The limit on a positive MTC ensures that core overheating accidents will not violate the accident analysis assumptions. The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions.

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement. The surveillance checks at BOC and MOC on an MTC provide confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

Typo

In MODE 1, the limits on the MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure startup and subcritical accidents, such as the uncontrolled CEA assembly or group withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES. However, the variation of the MTC, with temperature in MODES 3, 4, and 5, for DBAs

(continued)

BASES

APPLICABILITY
(continued) initiated in MODES 1 and 2, is accounted for in the subject accident analysis. The variation of the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC and MOC measurements are used for normalization.

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.4.1 and SR 3.1.4.2

The SRs for measurement of the MTC at the beginning and middle of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes smoothly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced to compensate for fuel depletion. The requirement for measurement prior to operation > 5% RTP satisfies the confirmatory check on the most positive (least negative) MTC value. The requirement for measurement, within 7 days after reaching 40 effective full power days and a ²/₃ core burnup, satisfies the confirmatory check of the most negative MTC value. The measurement is performed at any THERMAL POWER so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values may be extrapolated and compensated to permit direct comparison to the specified MTC limits.

SR 3.1.4.2 is modified by a Note that indicates performance is not required prior to entering MODE 1 or 2. Although this Surveillance is applicable in MODES 1 and 2, the

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.4.1 and SR 3.1.4.2 (continued)

reactor must be critical before the Surveillance can be completed. Therefore, entry into the applicable MODE prior to accomplishing the Surveillance is necessary.

SR 3.1.4.2 is modified by a second Note that indicates, if extrapolated MTC is more negative than the EOC COLR limit, the Surveillance may be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. An engineering evaluation is performed if the extrapolated value of MTC exceeds the Specification limits.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
 2. FSAR, Section [§].
 3. FSAR, Section [§].
 4. FSAR, Section [§].
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Element Assembly (CEA) Alignment ~~Original~~

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and regulating CEAs are initial assumptions in all safety analyses, which assume CEA insertion upon reactor trip. Maximum CEA misalignment is an initial assumption in the safety analyses which directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, GDC 26 (Ref. 1) and 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Plants (Ref. 2).

Mechanical or electrical failures may cause a CEA to become inoperable or to become misaligned from its group. CEA inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CEA worth for reactor shutdown. Therefore, CEA alignment and operability are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on CEA alignment and operability have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

CEAs are moved by their control element drive mechanisms (CEDMs). Each CEDM moves its CEA one step (approximately $\frac{3}{8}$ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Control Element Drive Mechanism Control System (CEDMCS).

The CEAs are arranged into groups that are radially symmetric. Therefore, movement of the CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating CEAs also provide reactivity (power level) control during normal operation and

(continued)

BASES

BACKGROUND (continued)

transients. Their movement may be automatically controlled by the Reactor Regulating System. Part-length CEAs are not credited in the safety analyses for shutting down the reactor, as are the regulating and shutdown groups. The part-length CEAs are used ~~solely~~ for ASI control *and for daily hand pulls*.

The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems, which are the Plant Computer CEA Position Indication System and the Reed Switch Position Indication System.

The Plant Computer CEA Position Indication System counts the commands sent to the CEA gripper coils from the CEDMCS that moves the CEAs. There is one step counter for each group of CEAs. Individual CEAs in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Plant Computer CEA Position Indication System is $\pm 2; 4, 2s$ considered highly precise (\pm one step or $\pm \frac{3}{4}$ inch). If a CEA does not move one step for each command signal, the step counter will still count the command and incorrectly reflect the position of the CEA.

The Reed Switch Position Indication System provides a highly accurate indication of actual CEA position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of reed switches spaced along a tube with a center to center distance of 1.5 inches, which is two steps. To increase the reliability of the system, there are redundant reed switches at each position.

APPLICABLE SAFETY ANALYSES

CEA misalignment accidents are analyzed in the safety analysis (Ref. 3). The accident analysis defines CEA misoperation as any event, with the exception of sequential group withdrawals, which could result from a single malfunction in the reactivity control systems. For example, CEA misalignment may be caused by a malfunction of the CEDM, CEDMCS, or by operator error. A stuck CEA may be caused by mechanical jamming of the CEA fingers or of the gripper. Inadvertent withdrawal of a single CEA may be caused by opening of the electrical circuit of the CEDM holding coil for a full-length or part-length CEA. A dropped CEA

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

✓ subgroup could be caused by an electrical failure in the CEA coil power programmers.

The acceptance criteria for addressing CEA inoperability or misalignment are that there be no violations of:

- a. Specified acceptable fuel design limits;
- b. Reactor Coolant System (RCS) pressure boundary integrity; and
- c. The core must remain subcritical after accident transients.

Three types of misalignment are distinguished. During movement of a group, one CEA may stop moving while the other CEAs in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one CEA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining CEAs to meet the SDM requirement with the maximum worth CEA stuck fully withdrawn. If a CEA is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck CEAs into account. The third type of misalignment occurs when one CEA drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).

Two types of analyses are performed in regard to static CEA misalignment (Ref. 4). With CEA banks at their insertion limits, one type of analysis considers the case when any one CEA is inserted [] inches into the core. The second type of analysis considers the case of a single CEA withdrawn [] inches from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio (DNBR) in both of these cases bounds the situation when a CEA is misaligned from its group by [7 inches].

Another type of misalignment occurs if one CEA fails to insert upon a reactor trip and remains stuck fully

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth CEA also fully withdrawn (Ref. 5).

✓ The effect of any misoperated CEA on the core power distribution will be assessed by the CEA calculators, and an appropriately augmented power distribution penalty factor will be supplied as input to the core protection calculators (CPCs). As the reactor core responds to the reactivity changes caused by the misoperated CEA and the ensuing reactor coolant and Doppler feedback effects, the CPCs will initiate a low DNBR or high local power density trip signal if specified acceptable fuel design limits (SAFDLs) are approached.

✓ Since the CEA drop incidents result in the most rapid approach to SAFDLs caused by a CEA misoperation, the accident analysis analyzed a single full length CEA drop, a single part length CEA drop, and a part length CEA subgroup drop. The most rapid approach to the DNBR SAFDL may be caused by either a single full length drop or a part length CEA subgroup drop depending upon initial conditions. The most rapid approach to the fuel centerline melt SAFDL is caused by a single part length CEA drop. *a full-strength CEA subgroup drop,*

✓ In the case of the full length CEA drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which when conservatively coupled result in local power and heat flux increases, and a decrease in DNBR. For plant operation within the DNBR and local power density (LPD) LCOs, DNBR and LPD trips can normally be avoided on a dropped CEA. *and a part-strength CEA drop*

✓ For a part length CEA subgroup drop, a distortion in power distribution, and a decrease in core power are produced. As the dropped part length CEA subgroup is detected, an appropriate power distribution penalty factor is supplied to the CPCs, and a reactor trip signal on low DNBR is generated. For the part length CEA drop, both core average power and three dimensional peak to average power density increase promptly. As the dropped part length CEA is detected, core power and an appropriately augmented power distribution penalty factor are supplied to the CPCs. *and a part-strength CEA subgroup drop*

CEA alignment satisfies Criteria 2 and 3 of the NRC Policy Statement.

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BASES (continued)

LCO

The limits on shutdown and regulating CEA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the CEAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the CEA banks maintain the correct power distribution and CEA alignment.

The requirement to maintain the CEA alignment to within [7 inches] between the highest and lowest CEAs in a subgroup is conservative. The minimum misalignment assumed in safety analysis is [19 inches], and in some cases, a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on CEA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (e.g., trippability) and alignment of CEAs have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the CEAs are bottomed, and the reactor is shut down and not producing fission power. In the shutdown modes, the OPERABILITY of the shutdown and regulating CEAs has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, SHUTDOWN MARGIN (SDM) ~~for SDM~~ for SDM in MODES 3, 4, and 5, and LCO 3.9.1, Boron Concentration, for boron concentration requirements during refueling.

ACTIONS

A.1, A.2.1, A.2.2, A.3.1, and A.3.2

A CEA may become misaligned, yet remain trippable. In this condition, the CEA can still perform its required function

(continued)

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.3.1, and A.3.2 (continued)

of adding negative reactivity should a reactor trip be necessary.

If one or more regulating CEAs are misaligned by [7 inches] and \leq [19 inches] but trippable, or one regulating CEA misaligned by $>$ [19 inches] but trippable, continued operation in MODES 1 and 2 may continue, provided, within 1 hour, the power is reduced in accordance with Figure 3.1.5-1, and SDM is \geq [52] ~~50~~ % $\Delta k/k$, and within 2 hours the misaligned CEA(s) is aligned within [7 inches] of its group or the misaligned CEA's group is aligned within [7 inches] of the misaligned CEA(s).

Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. Reducing THERMAL POWER in accordance with Figure 3.1.5-1 (in the accompanying LCO) ensures acceptable power distributions are maintained (Ref. 6). For small misalignments ($<$ [19 inches]) of the CEAs, there is:

- a. A small effect on the time dependent long term power distributions relative to those used in generating LCOs and limiting safety system settings (LSSS) setpoints;
- b. A small effect on the available SDM; and
- c. A small effect on the ejected CEA worth used in the accident analysis.

With a large CEA misalignment (\geq [19 inches]), however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on:

- a. The available SDM;
- b. The time dependent, long term power distributions relative to those used in generating LCOs and LSSS setpoints; and
- c. The ejected CEA worth used in the accident analysis.

(continued)

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.3.1, and A.3.2 (continued)

Therefore, this condition is limited to the single CEA misalignment, while still allowing 2 hours for recovery.

In both cases, a 2 hour time period is sufficient to:

- a. Identify cause of a misaligned CEA;
- b. Take appropriate corrective action to realign the CEAs; and
- c. Minimize the effects of xenon redistribution.

In this condition, an additional allowance must be made for the worth of the affected CEA when calculating the available SDM. With one or more misaligned CEAs, SDM must be verified for CEAs at the existing nonaligned positions. SDM is calculated by performing a reactivity balance calculation according to procedure, considering the listed effects in SR 3.1.1.1. This is necessary since the OPERABLE CEAs must still meet the single failure criterion. If additional negative reactivity is required to provide the necessary SDM, it must be provided by increasing the RCS boron concentration. One hour allows sufficient time to perform the SDM calculation and make any required boron adjustment to the RCS.

B.1, B.2.1, B.2.2, and B.3

If one or more shutdown CEAs are misaligned by $> [7 \text{ inches}]$ and $\leq [19 \text{ inches}]$ but trippable, or one shutdown CEA misaligned by $> [19 \text{ inches}]$ but trippable, continued operation in MODES 1 and 2 may continue, provided, within 1 hour, the power is reduced in accordance with Figure 3.1.5-1, and SDM is $\geq [5.0]\% \Delta k/k$, and within 2 hours the misaligned CEA(s) is aligned within $[7 \text{ inches}]$ of its group.

C.1, C.2.1, and C.2.2

If one or more part-length CEAs are misaligned by $> [7 \text{ inches}]$ and $\leq [19 \text{ inches}]$ or one part-length CEA misaligned by $> [19 \text{ inches}]$, continued operation in MODES 1 and 2 may continue, provided power is reduced in accordance

(continued)

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

with the appropriate figure within 1 hour, and within 2 hours the misaligned CEA(s) is restored to within [7 inches] of its group, or the misaligned CEA's group is aligned within [7 inches] of the misaligned CEA.

Although a partial length CEA has less of an effect on core flux than a full length CEA, a misaligned partial length CEA will still result in xenon redistribution and affect core power distribution. Requiring realignment within 2 hours minimizes these effects and ensures acceptable power distribution is maintained.

D.1

If a Required Action or associated Completion Time of Condition A, Condition B, or Condition C is not met, one or more regulating or shutdown CEAs are untrippable, or two or more CEAs are misaligned by > [19 inches], the unit is required to be brought to MODE 3. By being brought to MODE 3, the unit is brought outside its MODE of applicability.

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

If a CEA is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable CEA, meeting the insertion limits of LCO 3.1.6, Shutdown Control Element Assembly (CEA) Insertion Limits, and LCO 3.1.7, Regulating Control Element Assembly (CEA) Insertion Limits, does not ensure that adequate SDM exists. Therefore, the plant must be shut down in order to evaluate the SDM required boron concentration and power level for critical operation.

Continued operation is not allowed in the case of more than one CEA(s) misaligned from any other CEA in its group by > [19 inches], or with one or more full length CEAs untrippable. This is because these cases are indicative of

(continued)

BASES

ACTIONS

D.1 (continued)

a loss of SDM and power distribution, and a loss of safety function, respectively.

SURVEILLANCE
REQUIREMENTSSR 3.1.5.1

Verification that individual CEA positions are within [7 inches] (indicated reed switch positions) of all other CEAs in the group at a 12 hour Frequency allows the operator to detect a CEA that is beginning to deviate from its expected position. The specified Frequency takes into account other CEA position information that is continuously available to the operator in the control room, so that during actual CEA motion, deviations can immediately be detected.

SR 3.1.5.2

OPERABILITY of at least two CEA position indicator channels is required to determine CEA positions, and thereby ensure compliance with the CEA alignment and insertion limits. The CEA full in and full out limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions.

SR 3.1.5.3

Verifying each full length CEA is trippable would require that each CEA be tripped. In MODES 1 and 2 tripping each full length CEA would result in radial or axial power tilts, or oscillations. Therefore individual full length CEAs are exercised every 92 days to provide increased confidence that all full length CEAs continue to be trippable, even if they are not regularly tripped. A movement of [5 inches] is adequate to demonstrate motion without exceeding the alignment limit when only one full length CEA is being moved. The 92 day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the CEAs

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.5.3 (continued)

(Ref. 7). Between required performances of SR 3.1.5.3, if a CEA(s) is discovered to be immovable but remains trippable and aligned, the CEA is considered to be OPERABLE. At anytime, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of that CEA(s) must be made, and appropriate action taken.

SR 3.1.5.4

Performance of a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position over the entire Stroke Length of the CEA's travel. Since this test must be performed when the reactor is shut down, an 18 month Frequency to be coincident with refueling outage was selected. Operating experience has shown that these components usually pass this Surveillance when performed at a Frequency of once every 18 months. Furthermore, the Frequency takes into account other surveillances being performed at shorter Frequencies, which determine the OPERABILITY of the CEA Reed Switch Indication System.

SR 3.1.5.5

Verification of full Stroke Length CEA drop times determines that the maximum CEA drop time permitted is consistent with the assumed drop time used in the safety analysis (Ref. 7). Measuring drop times prior to reactor criticality, after reactor vessel head removal, ensures the reactor internals and CEDM will not interfere with CEA motion or drop time, and that no degradation in these systems has occurred that would adversely affect CEA motion or drop time. Individual CEAs whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is performed prior to criticality due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
 2. 10 CFR 50.46.
 3. FSAR, Section [15].
 4. FSAR, Section [15].
 5. FSAR, Section [15].
 6. FSAR, Section [15].
 7. FSAR, Section [15].
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Shutdown Control Element Assembly (CEA) Insertion Limits (~~Original~~)

BASES

BACKGROUND

The insertion limits of the shutdown CEAs are initial assumptions in all safety analyses which assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected CEA worth, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on shutdown CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected CEA worth, and SDM limits are preserved.

The shutdown CEAs are arranged into groups that are radially symmetric. Therefore, movement of the shutdown CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip.

The design calculations are performed with the assumption that the shutdown CEAs are withdrawn prior to the regulating CEAs. The shutdown CEAs can be fully withdrawn without the core going critical. This provides available negative reactivity for SDM in the event of boration errors. The shutdown CEAs are controlled manually or automatically by the control room operator. During normal unit operation, the shutdown CEAs are fully withdrawn. The shutdown CEAs must be completely withdrawn from the core prior to withdrawing regulating CEAs during an approach to criticality. The shutdown CEAs are then left in this position until the reactor is shut down. They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

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BASES (continued)

APPLICABLE
SAFETY ANALYSES

Accident analysis assumes that the shutdown CEAs are fully withdrawn any time the reactor is critical. This ensures that:

- a. The minimum SDM is maintained; and
- b. The potential effects of a CEA ejection accident are limited to acceptable limits.

CEAs are considered fully withdrawn at 145 inches, since this position places them outside the active region of the core.

On a reactor trip, all CEAs (shutdown CEAs and regulating CEAs), except the most reactive CEA, are assumed to insert into the core. The shutdown and regulating CEAs shall be at their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating CEAs may be partially inserted in the core as allowed by LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits." The shutdown CEA insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~21.1 > 20.0%~~") following a reactor trip from full power. The combination of regulating CEAs and shutdown CEAs (less the most reactive CEA, which is assumed to be fully withdrawn) is sufficient to take the reactor, from full power conditions at rated temperature to zero power, and to maintain the required SDM at ~~normal~~ load temperature (Ref. 3). The shutdown CEA insertion limit also limits the reactivity worth of ~~the~~ shutdown CEA.

The acceptance criteria for ~~the~~ shutdown CEA as well as regulating CEA insertion limits and inoperability or misalignment are that:

- a. There be no violation of:
 1. specified acceptable fuel design limits, or
 2. Reactor Coolant System pressure boundary damage integrity; and
- b. The core remains subcritical after accident transients.

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The shutdown CEA insertion limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The shutdown CEAs must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

APPLICABILITY

The shutdown CEAs must be within their insertion limits, with the reactor in MODES 1 and 2. The Applicability in MODE 2 begins any time any regulating CEA is not fully inserted. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODES 1 and 2, if shutdown CEAs are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation (considering the listed reactivity effects in Bases Section SR 3.1.1.1). In MODE 3, 4, 5, or 6, the shutdown CEAs are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1.1 ~~and LCO 3.1.1.2~~ for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.5.5, which verifies the freedom of the CEAs to move, and requires the shutdown CEAs to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1.1, A.1.2, and A.2

Prior to entering this Condition, the shutdown CEAs were fully withdrawn. If a shutdown CEA is then inserted into the core, its potential negative reactivity is added to the core as it is inserted. If boron concentration is not changed at this time, SDM should not change. This, however, is verified within 1 hour, or boration is initiated to bring

(continued)

BASES

ACTIONS

A.1.1, A.1.2, and A.2 (continued)

the SDM to within limit, if the CEA(s) is not restored to within limits prior to this time.

If the CEA(s) is not restored to within limits within 1 hour and the SDM is within limit, then an additional 1 hour is allowed for restoring the CEA(s) to within limits. The 2 hour total Completion Time allows the operator adequate time to adjust the CEA(s) in an orderly manner and is consistent with the required Completion Times in LCO 3.1.5, "Control Element Assembly (CEA) Alignment."

B.1

When Required Action A.1 or Required Action A.2 cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.6.1

Verification that the shutdown CEAs are within their insertion limits within 15 minutes prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown CEAs will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown CEAs are withdrawn before the regulating CEAs are withdrawn during a unit startup.

Since the shutdown CEAs are positioned manually by the control room operator, a verification of shutdown CEA position at a Frequency of 12 hours, is adequate to ensure that they are within their insertion limits. Also, the Frequency takes into account other information available to the operator in the control room for the purpose of monitoring the status of the shutdown CEAs.

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BASES (continued)

- REFERENCES
1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
 2. 10 CFR 50.46.
 3. FSAR, Section [15].
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Regulating Control Element Assembly (CEA) Insertion Limits *0010124*

BASES

BACKGROUND

The insertion limits of the regulating CEAs are initial assumptions in all safety analyses that assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available SDM, and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on regulating CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected CEA worth, reactivity insertion rate, and SDM limits are preserved.

The regulating CEA groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between CEA worth and position (integral CEA worth). The regulating CEA groups are withdrawn and operate in a predetermined sequence. The group sequence and overlap limits are specified in the COLR.

The regulating CEAs are used for precise reactivity control of the reactor. The positions of the regulating CEAs are manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.7; LCO 3.2.4, "Departure from Nucleate Boiling Ratio (DNBR)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)" provide limits on control component operation and on monitored process variables to ensure the core operates within LCO 3.2.1, "Linear Heat Rate (LHR)"; LCO 3.2.2, "Planar Radial Peaking Factor (F_{xy})"; and LCO 3.2.4, "Departure From Nucleate

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BASES

BACKGROUND
(continued)

Boiling Ratio (DNBR)" limits in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling System analysis. Operation within the F_{xy} and departure from nucleate boiling (DNB) limits given in the COLR prevents DNB during a loss of forced reactor coolant flow accident. In addition to the LHR, F_{xy} and DNBR limits, certain reactivity limits are preserved by regulating CEA insertion limits. The regulating CEA insertion limits also restrict the ejected CEA worth to the values assumed in the safety analyses and preserve the minimum required SDM in MODES 1 and 2.

The establishment of limiting safety system settings and LCOs require that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks, due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). From these analyses, CEA insertions are determined and a consistent set of radial peaking factors defined. The long term steady state and short term insertion limits are determined, based upon the assumed mode of operation used in the analyses, and provide a means of preserving the assumptions on CEA insertions used. The long and short term insertion limits of LCO 3.1.7 are specified for the plant, which has been designed for primarily base loaded operation, but has the ability to accommodate a limited amount of load maneuvering.

The regulating CEA insertion and alignment limits, ASI and T_q , are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the regulating bank insertion limits control the reactivity that could be added in the event of a CEA ejection accident, and the shutdown and

(continued)

BASES

BACKGROUND
(continued)

regulating bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow, ejected CEA, or other accident requiring termination by a Reactor Protection System trip function.

APPLICABLE
SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The acceptance criteria for the regulating CEA insertion, partial length CEA insertion, ASI, and T_q LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

- a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F. 10 CFR 50.46 (Ref. 2);
- b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel CEA in the core does not experience a DNB condition;
- c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and
- d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating CEA position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

increased power peaking and corresponding increased local LHRs.

The SDM requirement is ensured by limiting the regulating and shutdown CEA insertion limits, so that the allowable inserted worth of the CEAs is such that sufficient reactivity is available in the CEAs to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth CEA remains fully withdrawn upon trip (Ref. 4).

Operation at the insertion limits or ASI may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed T_b present. Operation at the insertion limit may also indicate the maximum ejected CEA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected CEA worths.

The regulating and shutdown CEA insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected CEA worth, and power distribution peaking factors are preserved (Ref. 5).

The regulating CEA insertion limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The limits on regulating CEA sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected CEA worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion and withdrawal, and is imposed to maintain acceptable power peaking during regulating CEA motion.

The power dependent insertion limit (PDIL) alarm circuit is required to be OPERABLE for notification that the CEAs are outside the required insertion limits. When the PDIL alarm circuit is inoperable, the verification of CEA positions is increased to ensure improper CEA alignment is identified before unacceptable flux distribution occurs.

(continued)

BASES (continued)

APPLICABILITY The regulating CEA sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution, ejected CEA worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected CEA worth assumptions would be exceeded in these MODES. SDM is preserved in MODES 3, 4, and 5 by adjustments to the soluble boron concentration.

This LCO is modified by a Note indicating the LCO requirement is suspended during SR 3.1.5.3. This SR verifies the freedom of the CEAs to move, and requires the regulating CEAs to move below the LCO limits, which would normally violate the LCO. The Note also allows the LCO to be not applicable during reactor power cutback operation, which inserts a selected CEA group (usually group 5) during loss of load events.

ACTIONS

A.1.1, A.1.2, A.2.1, and A.2.2

Operation beyond the transient insertion limit may result in a loss of SDM and excessive peaking factors. If the regulating CEA insertion limits are not met, then SDM must be verified by performing a reactivity balance calculation, considering the listed reactivity effects in Bases Section SR 3.1.1.1. One hour is sufficient time for conducting the calculation and commencing boration if the SDM is not within limits. The transient insertion limit should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the CEAs in response to changing plant conditions. When the regulating groups are inserted beyond the transient insertion limits, actions must be taken to either withdraw the regulating groups beyond the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual CEA insertion limit. Two hours provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

If the CEAs are inserted between the long term steady state insertion limits, the transient insertion limits for intervals > 4 hours per 24 hour period, and the short term steady state insertion limits are exceeded, peaking factors can develop that are of immediate concern (Ref. 6).

Additionally, since the CEAs can be in this condition without misalignment, penalty factors are not inserted in the core protection calculators (CPCs) to compensate for the developing peaking factors. Verifying the short term steady state insertion limits are not exceeded ensures that the peaking factors that do develop are within those allowed for continued operation. Fifteen minutes provides adequate time for the operator to verify if the short term steady state insertion limits are exceeded.

Experience has shown that rapid power increases in areas of the core, in which the flux has been depressed, can result in fuel damage as the LHR in those areas rapidly increases. Restricting the rate of THERMAL POWER increases to $\leq 5\%$ RTP per hour, following CEA insertion beyond the long term steady state insertion limits, ensures the power transients experienced by the fuel will not result in fuel failure (Ref. 7).

C.1

With the regulating CEAs inserted between the long term steady state insertion limit and the transient insertion limit, and with the core approaching the 5 effective full power days (EFPD) per 30-EFPD, or 14-EFPD per 365-EFPD limits, the core approaches the acceptable limits placed on operation with flux patterns outside those assumed in the long term burnup assumptions. In this case, the CEAs must be returned to within the long term steady state insertion limits, or the core must be placed in a condition in which the abnormal fuel burnup can not continue. A Completion Time of 2 hours is a reasonable time to return the CEAs to within the long term steady state insertion limits.

The required Completion Time of 2 hours from initial discovery of a regulating CEA group outside the limits until its restoration to within the long term steady state limits.

(continued)

BASES

ACTIONS

C.1 (continued)

shown on the figures in the COLR, allows sufficient time for borated water to enter the Reactor Coolant System from the chemical addition and makeup systems, and to cause the regulating CEAs to withdraw to the acceptable region. It is reasonable to continue operation for 2 hours after it is discovered that the 5 or 14 day EFPD limit has been exceeded. This Completion Time is based on limiting the potential xenon redistribution, the low probability of an accident, and the steps required to complete the action.

D.1.1, D.1.2, D.2.1, and D.2.2

If the regulating CEA insertion limits are not met, then SDM must be verified by performing a reactivity balance calculation, considering the effects in SR 3.1.1.1 bases. One hour is sufficient time for conducting the calculation and commencing boration if the SDM is not within limits.

With the Core Operating Limit Supervisory System out of service, operation beyond the short term steady state insertion limits can result in peaking factors that could approach the DNB or local power density trip setpoints. Eliminating this condition within 2 hours limits the magnitude of the peaking factors to acceptable levels (Ref. 8). Restoring the CEAs to within the limit or reducing THERMAL POWER to that fraction of RTP that is allowed by CEA group position, using the limits specified in the COLR, ensures acceptable peaking factors are maintained.

E.1

With the PDIL circuit inoperable, performing SR 3.1.7.1 within 1 hour and every 4 hours thereafter ensures improper CEA alignments are identified before unacceptable flux distributions occur.

F.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is

(continued)

BASES

ACTIONS

E.1 (continued)

reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.1

With the PDIL alarm circuit OPERABLE, verification of each regulating CEA group position every 12 hours is sufficient to detect CEA positions that may approach the acceptable limits, and provide the operator with time to undertake the Required Action(s), should the sequence or insertion limits be found to be exceeded. The 12 hour Frequency also takes into account the indication provided by the PDIL alarm circuit and other information about CEA group positions available to the operator in the control room.

SR 3.1.7.1 is modified by a Note indicating that entry is allowed into MODE 2 without having performed the SR. This is necessary, since the unit must be in the applicable MODES in order to perform Surveillances that demonstrate the LCO limits are met.

SR 3.1.7.2

Verification of the accumulated time of CEA group insertion between the long term steady state insertion limits and the transient insertion limits ensures the cumulative time limits are not exceeded. The 24 hour Frequency ensures the operator identifies a time limit that is being approached before it is reached.

SR 3.1.7.3

Demonstrating the PDIL alarm circuit OPERABLE verifies that the PDIL alarm circuit is functional. The 31 day Frequency takes into account other Surveillances being performed at shorter Frequencies that identify improper CEA alignments.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
 2. 10 CFR 50.46.
 3. FSAR, Section [15], Section [17], and Section [18].
 4. FSAR, Section [17].
 5. FSAR, Section [15].
 6. FSAR, Section [15].
 7. FSAR, Section [16].
 8. FSAR, Section [15].
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Part Length Control Element Assembly (CEA) Insertion Limits ~~(Optional)~~ ~~(Digital)~~

BASES

BACKGROUND

The insertion limits of the ~~part~~ ^{core} length CEAs are initial assumptions in all safety analyses. The insertion limits directly affect core power distributions. The applicable criteria for these power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Plants" (Ref. 2). Limits on ~~part~~ ^{core} length CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution defined by the design power peaking limits is preserved.

The regulating CEAs are used for precise reactivity control of the reactor. The positions of the regulating CEAs are manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits"; LCO 3.1.8; LCO 3.2.4, "Departure From Nucleate Boiling Ratio (DNBR)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LHR) (LCO 3.2.1, "Linear Heat Rate (LHR)"); planar peaking factor (F_{xy}) (LCO 3.2.2, "Planar Radial Peaking Factors (F_{xy})"); and LCO 3.2.4 limits in the COLR.

Operation within the limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling System analysis. Operation within the F_{xy} and departure from nucleate boiling (DNB) limits given in the COLR prevents DNB during a loss of forced reactor coolant flow accident.

The establishment of limiting safety system settings and LCOs requires that the expected long and short term

(continued)

BASES

BACKGROUND (continued)

behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup; it is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). From these analyses, CEA insertions are determined, and a consistent set of radial peaking factors are defined. The long term (steady state) and short term insertion limits are determined, based upon the assumed mode of operation used in the analyses; they provide a means of preserving the assumptions on CEA insertions used. The long and short term insertion limits of LCO 3.1.8 are specified for the plant, which has been designed primarily for base loaded operation, but has the ability to accommodate a limited amount of load maneuvering.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The regulating CEA insertion, part ^{Still} length CEA insertion, ASI, and T₀ LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

- a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2);
- b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel CEA in the core does not experience a DNB condition;
- c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating CEA position, part ~~Length~~ CEA position, ASI, and T_a are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The regulating CEA insertion limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The limits on part ~~Length~~ CEA insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution.

APPLICABILITY

The part ~~Length~~ insertion limits shall be maintained with the reactor in MODE 1 > 20% RTP. These limits must be maintained, since they preserve the assumed power distribution. Applicability in lower MODES is not required, since the power distribution assumptions would not be exceeded in these MODES.

This LCO has been modified by a Note suspending the LCO requirement while exercising part ~~Length~~ CEAs. Exercising part ~~Length~~ CEAs may require moving them outside their insertion limits.

ACTIONS

A.1, A.2, and B.1

If the part ~~Length~~ CEA groups are inserted beyond the transient insertion limit or between the long term

(continued)

BASES

ACTIONS

A.1, A.2, and B.1 (continued)

(steady state) insertion limit and the transient limit for 7 or more effective full power days (EFPD) out of any 30-EFPD period, or for 14-EFPD or more out of any 365-EFPD period, flux patterns begin to develop that are outside the range assumed for long term fuel burnup. If allowed to continue beyond this limit, the peaking factors assumed as initial conditions in the accident analysis may be invalidated (Ref. 4). Restoring the CEAs to within limits or reducing THERMAL POWER to that fraction of RTP that is allowed by CEA group position, using the limits specified in the COLR, ensures that acceptable peaking factors are maintained.

Since these effects are cumulative, actions are provided to limit the total time the part Length CEAs can be out of limits in any 30-EFPD or 365-EFPD period. Since the cumulative out of limit times are in days, an additional Completion Time of 2 hours is reasonable for restoring the part Length CEAs to within the allowed limits.

C.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should commence. A Completion Time of 4 hours is reasonable, based on operating experience, for reducing power to ≤ 20 RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.8.1

Verification of each part Length CEA group position every 12 hours is sufficient to detect CEA positions that may approach the limits, and provide the operator with time to undertake the Required Action(s), should insertion limits be found to be exceeded. The 12 hour Frequency also takes into account the indication provided by the power dependent insertion limit alarm circuit and other information about CEA group positions available to the operator in the control room.

(continued)

BASES (continued)

- REFERENCES
1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
 2. 10 CFR 50.46.
 3. FSAR, Section [15].
 4. FSAR, Section [16].
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 Special Test Exceptions (STE) - SHUTDOWN MARGIN (SDM) ~~(Digital)~~

BASES

BACKGROUND

The primary purpose of the SDM STE is to permit relaxation of existing LCDs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine the control element assembly (CEA) worth and SDM.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

The key objectives of a test program are to (Ref. 3):

- a. Ensure that the facility has been adequately designed;
- b. Validate the analytical models used in design and analysis;
- c. Verify assumptions used for predicting plant response;
- d. Ensure that installation of equipment in the facility has been accomplished, in accordance with the design; and
- e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics

(continued)

BASES

BACKGROUND
(continued)

of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved, in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE
SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because adequate limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). PHYSICS TESTS for reload fuel cycles are given in Table 1 of ANSI/ANS-19.6.1-1985. Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate (LHR) remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

- a. LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~4.1.1.1~~": and
- b. LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits."

Therefore, this LCO places limits on the minimum amount of CEA worth required to be available for reactivity control when CEA worth measurements are performed.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The individual LCOs cited above govern SDM CEA group height, insertion, and alignment. Additionally, the LCOs governing Reactor Coolant System (RCS) flow, reactor inlet temperature T_{ci} , and pressurizer pressure contribute to maintaining departure from nucleate boiling (DNB) parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the loss of coolant accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 6). The criteria for the loss of forced reactor coolant flow accidents are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

SRs are conducted as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these SRs allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

Requiring that shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) be available for trip insertion from the OPERABLE CEAs, provides a high degree of assurance that shutdown capability is maintained for the most challenging postulated accident, a stuck CEA. Since LCO 3.1.1 is suspended, however, there is not the same degree of assurance during this test that the reactor would always be shut down if the highest worth CEA was stuck out and calculational uncertainties or the estimated highest CEA worth was not as expected (the single failure criterion is not met). This situation is judged acceptable, however, because specified acceptable fuel damage limits are still met. The risk of experiencing a stuck CEA and subsequent criticality is reduced during this PHYSICS TEST exception by the requirements to determine CEA positions every 2 hours; by the trip of each CEA to be withdrawn within 24 hours prior to suspending the SDM; and by ensuring that shutdown reactivity is available, equivalent to the reactivity worth of the estimated highest worth withdrawn CEA (Ref. 5).

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are total planar radial peaking factor, total integrated radial

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

peaking factor, T_d , and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

LCO

This LCO provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This STE is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations. The SDM requirements of LCO 3.1.1 and the regulating CEA insertion limits of LCO 3.1.7 may be suspended.

APPLICABILITY

This LCO is applicable in MODES 2 and 3. Although CEA worth testing is conducted in MODE 2, sufficient negative reactivity is inserted during the performance of these tests to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the STE allows limited operation to 6 consecutive hours in MODE 3 as indicated by the Note, without having to borate to meet the SDM requirements of LCO 3.1.1.

ACTIONS

A.1

With any CEA not fully inserted and less than the minimum required reactivity equivalent available for insertion, or with all CEAs inserted and the reactor subcritical by less than the reactivity equivalent of the highest worth withdrawn CEA, restoration of the minimum SDM requirements must be accomplished by increasing the RCS boron concentration. The required Completion Time of 15 minutes

(continued)

BASES

ACTIONS

A.1 (continued)

for initiating boration allows the operator sufficient time to align the valves and start the boric acid pumps, and is consistent with the Completion Time of LCO 3.1.1.

SURVEILLANCE
REQUIREMENTSSR 3.1.9.1

Verification of the position of each partially or fully withdrawn full ~~length~~ or part ~~length~~ CEA is necessary to ensure that the minimum negative reactivity requirements for insertion on a trip are preserved. A 2 hour Frequency is sufficient for the operator to verify that each CEA position is within the acceptance criteria.

SR 3.1.9.2

Prior demonstration that each CEA to be withdrawn from the core during PHYSICS TESTS is capable of full insertion, when tripped from at least a 50% withdrawn position, ensures that the CEA will insert on a trip signal. The [7 day] Frequency ensures that the CEAs are OPERABLE prior to reducing SDM to less than the limits of LCO 3.1.1.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
 2. 10 CFR 50.59.
 3. Regulatory Guide 1.68, Revision 2, August 1978.
 4. ANSI/ANS-19.6.1-1985, December 13, 1985.
 5. FSAR, Chapter 14.
 6. 10 CFR 50.46.
 7. FSAR, Chapter 15.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.10 Special Test Exceptions (STE) - MODES 1 and 2 ~~(Original)~~

BASES

BACKGROUND

The primary purpose of these MODES 1 and 2 STEs is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine specific reactor core characteristics.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

The key objectives of a test program are to (Ref. 3):

- a. Ensure that the facility has been adequately designed;
- b. Validate the analytical models used in design and analysis;
- c. Verify assumptions used for predicting plant response;
- d. Ensure that installation of equipment in the facility has been accomplished, in accordance with design; and
- e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

(continued)

BASES

BACKGROUND (continued)

PHYSICS TESTS procedures are written and approved, in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate (LHR) remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

- LCO 3.1.4. "Moderator Temperature Coefficient (MTC)";
- LCO 3.1.5. "Control Element Assembly (CEA) Alignment";
- LCO 3.1.6. "Shutdown Control Element Assembly (CEA) Insertion Limits";
- LCO 3.1.7. "Regulating Control Element Assembly (CEA) Insertion Limits (F_{xy}^T)";
- LCO 3.1.8. "Partial Length Control Element Assembly (CEA) Insertion Limits";
- LCO 3.2.2. "Planar Radial Peaking Factors"; and
- LCO 3.2.3. "AZIMUTHAL POWER TILT (T_q)."

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

REPRESENT CONSERVATIVE
VALUES THAT, BASED ON
SAFETY ANALYSES AND ON
OPERATING EXPERIENCE
PERFORMING PHYSICS
TESTS, ENSURE THAT
LHRs ARE MAINTAINED
WITHIN ACCEPTABLE
LIMITS.

The safety analysis (Ref. 7) places limits on allowable THERMAL POWER during PHYSICS TESTS and requires that the LHR and the departure from nucleate boiling (DNB) parameter be maintained within limits. The power plateau of $< 85\%$ RTP and the associated trip setpoints are required to ensure ~~explain~~. SDM shall be maintained $\geq []\% \Delta k/k$.

The individual LCOs governing CEA group height, insertion and alignment, ASI, total planar radial peaking factor, total integrated radial peaking factor, and T_a , preserve the LHR limits. Additionally, the LCOs governing Reactor Coolant System (RCS) flow, reactor inlet temperature (T_c), and pressurizer pressure contribute to maintaining DNB parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the loss of coolant accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 6). The criteria for the loss of forced reactor coolant flow accident are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR and DNB parameter limits may be suspended. The results of the accident analysis are not adversely impacted, however, if LHR and DNB parameters are verified to be within their limits while the LCOs are suspended. Therefore, SRs are placed as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these Surveillances allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are total planar radial peaking factor, total integrated radial peaking factor, T_a , and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the component and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

LCO

This LCO permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of PHYSICS TESTS, such as those required to:

- a. Measure CEA worth;
- b. Determine the reactor stability index and damping factor under xenon oscillation conditions;
- c. Determine power distributions for non normal CEA configurations;
- d. Measure rod shadowing factors; and
- e. Measure temperature and power coefficients.

Additionally, it permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient (ITC), MTC, and power coefficient.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, LCO 3.1.8, LCO 3.2.2, and LCO 3.2.3 may be suspended during the performance of PHYSICS TESTS provided:

- a. THERMAL POWER is restricted to test power plateau, which shall not exceed 85% RTP; and
- b. SDM shall be $\geq [4.0]\% \Delta k/k$.

6.5

APPLICABILITY

This LCO is applicable in MODES 1 and 2 because the reactor must be critical at various THERMAL POWER levels to perform the PHYSICS TESTS described in the LCO section. Limiting the test power plateau to < 85% RTP ensures that LHRs are maintained within acceptable limits.

(continued)

BASES (continued)

ACTIONS

A.1

If THERMAL POWER exceeds the test power plateau in MODE 1, THERMAL POWER must be reduced to restore the additional thermal margin provided by the reduction. The 15 minute Completion Time ensures that prompt action shall be taken to reduce THERMAL POWER to within acceptable limits.

B.1 and B.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until the SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

C.1 and C.2

If Required Actions A.1 or B.1 cannot be completed within the required Completion Time, PHYSICS TESTS must be suspended within 1 hour, and the reactor must be brought to MODE 3. Allowing 1 hour for suspending PHYSICS TESTS allows the operator sufficient time to change any abnormal CEA configuration back to within the limits of LCO 3.1.5, LCO 3.1.6, and LCO 3.1.7. Bringing the reactor to MODE 3 within 6 hours increases thermal margin and is consistent with the Required Actions of the power distribution LCOs. The required Completion Time of 6 hours is adequate for performing a controlled shutdown from full power conditions in an orderly manner and without challenging plant systems, and is consistent with the power distribution LCO Completion Times.

SURVEILLANCE
REQUIREMENTSSR 3.1.10.1

Verifying that THERMAL POWER is equal to or less than that allowed by the test power plateau, as specified in the

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.10.1 (continued)

PHYSICS TEST procedure and required by the safety analysis, ensures that adequate LHR and departure from nucleate boiling ratio margins are maintained while LCOs are suspended. The 1 hour Frequency is sufficient, based upon the slow rate of power change and increased operational controls in place during PHYSICS TESTS. Monitoring LHR ensures that the limits are not exceeded.

SR 3.1.10.2

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

- a. RCS boron concentration;
- b. Control bank position;
- c. RCS average temperature;
- d. Fuel burnup based on gross thermal energy generation;
- e. Xenon concentration; and
- f. ITC.

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. Regulatory Guide 1.68, Revision 2, August 1978.
4. ANSI/ANS-19.6.1-1985, December 13, 1985.

(continued)

BASES

REFERENCES
(continued)

5. FSAR, Chapter [14].
 6. 10 CFR 50.46.
 7. FSAR, Section [15.3.2.1].
-

ADD BASES

FOR NEW 3.1.11

& 3.1.12 SECTIONS

AS SHOWN ON

ATTACHED PAGES.

16A.4.7

¹¹
B 3.1.40 SPECIAL TEST EXCEPTIONS - CEDMS TESTING

Special Test Exception - CEDMS Testing

B 3.1.40

11

B 3.1 REACTIVITY CONTROL SYSTEMS

¹¹
B 3.1.40 Special Test Exception - CEDMS Testing

BASES

BACKGROUND

CEDMS testing is performed startup to verify the operability of the control element drives. Since this test requires the withdrawal of CEAs, the shutdown margin is reduced. In order that the test may be performed, this special test exception is provided since the requirements of LCO 3.1.1 would be too restrictive to allow performance of the test.

APPLICABLE
SAFETY ANALYSIS

In Ref. 1, the conditions of the CEDMCS testing were analyzed. It was found that sufficient subcriticality is maintained in case of a CEA ejection accident. This is from the fact that prior to testing $K(n-1)$ must be less than 0.99. The margin will preclude inadvertent criticality.

LCO

Suspension of the shutdown margin requirement of LCO 3.1.1 may be suspended for pre-startup testing of the CEDMS if four conditions are met. First, only one CEA may be withdrawn at a time. Second, no CEA may be withdrawn more than seven inches. Third, with RTCBs open, $K(n-1)$ must be less than 0.99 before the start of testing. Fourth, all other operations which involve a reactivity increase must be suspended during testing.

APPLICABILITY

¹¹
LCO 3.1.40 is applicable during MODES 4 and 5 since these are the modes during which CEDMS testing is performed.

ACTIONS

A.1

If any of the four requirements are not met then testing must be suspended and the shutdown margin must be restored to the limit of LCO 3.1.1. This action is necessary for the prevention of an inadvertent criticality.

(continued)

Special Test Exception - CEDMS Testing

B 3.1.1~~2~~

1/

BASES

SURVEILLANCE
REQUIREMENTS

1/1
SR 3.1.1~~2~~.1

Determination of the shutdown margin ensures that CEDMS testing is being performed under conditions that would prevent an inadvertent criticality. The frequency of 24 hours is based upon operating experience and the fact that other administrative controls exist to prevent unauthorized reactivity increases.

REFERENCES

1. Safety Evaluation by the Office of NRR, Docket no. STN 50-530, January 26, 1988.
2. CESSAR-DC, Section 19.8, "Shutdown Risk Assessment".

16A.4.8 ¹² B 3.1.4 BORON DILUTION ALARMS

Boron Dilution Alarms

B 3.1.4

¹²

B 3.1 REACTIVITY CONTROL SYSTEMS

¹² B 3.1.4 Boron Dilution Alarms

BASES

BACKGROUND There are two startup channel high neutron flux alarms in the System 80+ design. These alarms exist for the purpose of alerting the operator to an inadvertent boron dilution and subsequently the prevention of an inadvertent criticality.

APPLICABLE SAFETY ANALYSIS In Ref. 1, it is mentioned that the use of two high neutron flux alarms provide proper redundancy for the detection of a boron dilution event. A single alarm failure will still leave the operator with adequate high neutron flux detection capability.

LCO The LCO requires that both startup channel high neutron flux alarms shall be operable.

APPLICABILITY ¹² LCO 3.1.4 is applicable during MODES 3,4,5, and 6. Since the reactor is critical in MODE 1 and also is critical (or approaching critical) in MODE 2, this LCO does not apply in MODES 1 and 2.

ACTION A.1 and A.2

With one startup channel high neutron flux alarm inoperable, action must be immediately taken to restore the inoperable channel to operable status. Also the RCS boron concentration must be determined when entering MODE 3, 4, 5, or 6 or at the time the alarm is determined inoperable. This second action is to be performed immediately and once per the frequency given in the LCO tables 3.1.4-1 through 3.1.4-5. These actions ensure that an alternate means is available for the detection of an inadvertent boron dilution event.

(continued)

Boron Dilution Alarms

B 3.1. ~~1~~

12

BASES

ACTION
(continued)

B.1, B.2, and B.3

With both startup channel high neutron flux alarms inoperable, action must be immediately taken to restore a single channel to operable status. Also the RCS boron concentration must be determined when entering MODE 3, 4, 5, 6 or at the time of alarm is determined inoperable. This second action is to be performed immediately ~~and~~ once per the frequency given in the LCO Tables 3.1. ~~1~~-1 through 3.1. ~~1~~-5. Immediate suspension of all operations (continued) involving core alterations or positive reactivity changes is also required. These actions will help prevent the loss of shutdown margin and return to criticality should an inadvertent boron dilution occur.

SURVEILLANCE

12

SR 3.1. ~~1~~.1

A channel check shall be performed on each startup channel once per 12 hours to ensure proper operation. The frequency is based upon operating experience and administrative controls.

12

SR 3.1. ~~1~~.2

A channel calibration shall be performed on each startup channel every 31 days of cumulative operation during shutdown. The frequency is based upon operating experiences.

REFERENCES

1. PVNGS Technical Specification Bases, 3/4.1.2.7.
2. CESSAR-DC, Section 19.8, "Shutdown Risk Assessment".

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Linear Heat Rate (LHR) ~~Digital~~

LCO 3.2.1 LHR shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core Operating Limit Supervisory System (COLSS) calculated core power exceeds the COLSS calculated core power operating limit based on LHR.	A.1 Restore LHR to within limits.	1 hour
B. LHR not within region of acceptable operation when the COLSS is out of service.	B.1 Restore LHR to within limits.	4 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to \leq 20% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.1.1 -----NOTE----- Only applicable when COLSS is out of service. With COLSS in service, LHR is continuously monitored. ----- Verify LHR, as indicated on each OPERABLE local power density channels, is $\leq [13.7 \text{ kW/ft}]$.</p>	<p>2 hours</p>
<p>SR 3.2.1.2 Verify the COLSS margin alarm accuates at a THERMAL POWER equal to or less than the core power operating limit based on LHR.</p>	<p>31 days</p>

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Planar Radial Peaking Factors (F_{xy}) (~~Digital~~)

LC0 3.2.2 The measured Planar Radial Peaking Factors (F_{xy}^m) shall be equal to or less than the Planar Radial Peaking Factors (F_{xy}^c). (These factors are used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPCs)).

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F_{xy}^m > F_{xy}^c$	A.1.1 Adjust addressable CPC constants to increase the multiplier applied to planar radial peaking by a factor $\geq F_{xy}^m/F_{xy}^c$.	6 hours
	AND	
	A.1.2 Maintain a margin to the COLSS operating limits of $[(F_{xy}^m/F_{xy}^c)-1.0] \times 100\%$.	6 hours
	OR	
	A.2 Adjust the affected F_{xy}^c used in the COLSS and CPCs to a value greater than or equal to the measured F_{xy}^m .	6 hours
	OR	
	A.3 Reduce THERMAL POWER to $\leq 20\%$ RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.2.1 Verify measured F_{xy}^m obtained using the Incore Detector System is equal to or less than the value of F_{xy}^e used in the COLSS and CPCs.</p>	<p>Once after each fuel loading with THERMAL POWER > 40% RTP but prior to operations above 70% RTP</p> <p><u>AND</u></p> <p>31 EFPD thereafter</p>

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AZIMUTHAL POWER TILT (T_q) (~~Digital~~)

LC0 3.2.3 The measured T_q shall be less than or equal to the T_q allowance used in the core protection calculators (CPCs).

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured T _q greater than the allowance used in the CPCs and ≤ 0.10.	A.1 Restore measured T _q .	2 hours
	<u>OR</u> A.2 Adjust the T _q allowance in the CPCs to greater than or equal to the measured value.	2 hours
B. Measured T _q > 0.10.	-----NOTE----- All subsequent Required Actions must be completed if power reduction commences prior to restoring T _q to ≤ 0.10. -----	
	B.1 Reduce THERMAL POWER to ≤ 50% RTP. <u>AND</u>	4 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Reduce Linear Power Level-High trip setpoints to $\leq 55\%$ RTP.	16 hours
	AND B.3 Restore the measured T _a to less than the T _a allowance used in the CPCs.	Prior to increasing THERMAL POWER -----NOTE----- Correct the cause of the out of limit condition prior to increasing THERMAL POWER. Subsequent power operation > 50% RTP may proceed provided that the measured T _a is verified ≤ 0.10 at least once per hour for 12 hours, or until verified at $\geq 95\%$ RTP -----
C. Required Actions and associated Completion Times not met.	C.1 Reduce THERMAL POWER to $\leq 20\%$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.3.1</p> <p>-----NOTES----- Only applicable when COLSS is out of service. With COLSS in service, this parameter is continuously monitored. -----</p> <p>Calculate T_a and verify it is within the limit.</p>	<p>12 hours</p>
<p>SR 3.2.3.2</p> <p>Verify COLSS azimuthal tilt alarm is actuated at a T_a value less than the T_a value used in the CPCs.</p>	<p>31 days</p>
<p>SR 3.2.3.3</p> <p>Independently confirm the validity of the COLSS calculated T_a by use of the incore detectors.</p>	<p>31 EFPD (days)</p>

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Departure From Nucleate Boiling Ratio (DNBR) ~~3.2.4~~

LCO 3.2.4

The DNBR shall be maintained by one of the following methods:

- a. Maintaining Core Operating Limit Supervisory System (COLSS) calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and either one or both control element assembly calculators (CEACs) are OPERABLE);
- b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by 13.0% RTP (when COLSS is in service and neither CEAC is OPERABLE);
- c. Operating within the region of acceptable operation of Figure 3.2.4-1 specified in the COLR using any operable core protection calculator (CPC) channel (when COLSS is out of service and either one or both CEACs are OPERABLE); or
- d. Operating within the region of acceptable operation of Figure 3.2.4-2 specified in the COLR using any operable CPC channel (when COLSS is out of service and neither CEAC is OPERABLE).

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. COLSS calculated core power not within limit.	A.1 Restore the DNBR to within limit.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. DNBR outside the region of acceptable operation when COLSS is out of service.	B.1 Restore DNBR to within limit.	4 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to $\leq 20\%$ RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.4.1 -----NOTE----- Only applicable when COLSS is out of service. With COLSS in service, this parameter is continuously monitored. -----</p> <p>Verify DNBR, as indicated on all OPERABLE DNBR channels, is within the limit of Figure 3.2.4-1 or 3.2.4-2 of the COLR, as applicable.</p>	2 hours
SR 3.2.4.2 Verify COLSS margin alarm actuates at a THERMAL POWER level equal to or less than the core power operating limit based on DNBR.	31 days

3.2 POWER DISTRIBUTION LIMITS

3.2.5 AXIAL SHAPE INDEX (ASI) ~~Digital~~

LC0 3.2.5 ASI shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core average ASI not within limits.	A.1 Restore ASI to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to \leq 20% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.5.1 Verify ASI is within limits.	12 hours

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Linear Heat Rate (LHR) (Digital) →

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

- a. Using full or part ^{core} length CEAs to alter the axial power distribution;
- b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and
- c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

(continued)

BASES

BACKGROUND
(continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the LHR and departure from nucleate boiling (DNB).

[1.24] ad

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bow and grid spacers. It is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) and the core protection calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and the DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating an actual value of DNBR and local power density (LPD) for comparison with the respective trip setpoints.

A DNBR penalty factor is included in both the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience a greater magnitude of rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

(continued)

BASES

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded during AOOs by initiating reactor trips.

The COLSS continually generates an assessment of the calculated margin for specified LHR and DNBR limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicate CEA positions. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high LPD or low DNBR trips in the RPS initiate a reactor trip prior to the exceeding of fuel design limits.

The LHR and DNBR algorithms are valid within the limits on ASI, F_{xy} and T_a . These limits are obtained directly from initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4).

The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

- a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

INDENT

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. ~~ALA~~); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (~~SDR/264~~) Ref. ~~ALA~~.

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing the LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_a limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core.

Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Fuel cladding damage does not occur from conditions outside the limits of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs from initial conditions outside the limits of these LCOs. This

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and can correspondingly increase local LHR.

The LHR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR. The limitation on LHR ensures that in the event of a LOCA the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

- a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate due to the poor signal to noise ratios at relatively low core power levels; and
 - b. As a result of this inaccuracy, the CPCs assume minimum core power of 20% RTP when generating LPD and DNBR trip signals. When core power is below 20% RTP, the core is operating well below its thermal limits and the resultant CPC calculated LPD and DNBR trips are highly conservative.
-

ACTIONS

A.1

Operation at or below the COLSS calculated power limit based on the LHR ensures that the LHR limit is not exceeded. If the COLSS calculated core power limit based on the LHR exceeds the operating limit, restoring the LHR to within limit in 1 hour ensures that prompt action is taken to reduce LHR to below the specified limit. One hour is a reasonable time to return LHR to within limits when the

(continued)

BASES

ACTIONS

A.1 (continued)

limit is exceeded without a trip due to events such as a dropped CEA or an axial xenon oscillation.

B.1

If the COLSS is not available the OPERABLE LPD channels are monitored to ensure that the LHR limit is not exceeded. Operation within this limit ensures that in the event of a LOCA the fuel cladding temperature does not exceed 2200°F. Four hours is allowed for restoring the LHR limit to within the region of acceptable operation. This duration is reasonable because the COLSS allows the plant to operate with less LHR margin (closer to the LHR limit than when monitoring the CPCs).

Also, when operating with the COLSS out of service there is a possibility of a slow undetectable transient that degrades the LHR slowly over the 4 hour period and is then followed by an AOD or an accident. To remedy this, the CPC calculated values of LHR are monitored every 15 minutes when the COLSS is out of service. Also, a maximum allowable change in the CPC calculated LHR ensures that further degradation requires the operators to take immediate action to reduce reactor power to comply with the Technical Specifications (TS). Implementation of this procedure ensures that reductions in core thermal margin are quickly detected, and if necessary, results in a decrease in reactor power and subsequent compliance with the existing COLSS out of service TS limits.

Four hours is allowed to restore the LHR to within limits if the COLSS is not restored to OPERABLE status. This duration is reasonable because the Frequency of the CPC determination of LHR is increased but, with the operation maintained steady, the likelihood of exceeding the LHR limit during the additional 2 hours is not increased. Also, the likelihood of induced reactor transients from an early power reduction is decreased during this period.

which?
ADDL
2 hrs
B?

to the next surveillance

(continued)

BASES

ACTIONS
(continued)

C.1

If the LHR cannot be returned to within its limit or the LHR cannot be determined because of the COLSS and CPC inoperability, core power must be reduced. Reduction of core power to < 20% RTP ensures that the core is operating within its thermal limits and places the core in a conservative condition based on the trip setpoints generated by the CPCs, which assume a minimum core power of 20% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 20% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.1

With the COLSS out of service, the operator must monitor the LHR with each OPERABLE local power density channel. A 2 hour Frequency is sufficient to allow the operator to identify trends that would result in an approach to the LHR limits.

This SR is modified by a Note that states that the SR is applicable only when the COLSS is out of service. Continuous monitoring of the LHR is provided by the COLSS, which calculates core power and core power operating limits based on the LHR and continuously displays these limits to the operator. A COLSS margin alarm is annunciated in the event that the THERMAL POWER exceeds the core power operating limit based on LHR.

SR 3.2.1.2

Verification that the COLSS margin alarm actuates at a THERMAL POWER level equal to or less than the core power operating limit based on the LHR in units of kilowatts per foot ensures the operator is alerted when conditions approach the LHR operating limit.

The 31 day Frequency for performance of this SR is consistent with the historical testing frequency of reactor protection and monitoring systems. The Surveillance Frequency for testing protection systems was extended to

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.2 (continued)

92 days by CEN 327. Monitoring systems were not addressed in CEN 327, therefore this Frequency remains at 31 days.

REFERENCES

1. FSAR, Section [15].
2. FSAR, Section [6].
3. CE-1 Correlation for DNBR.
4. 10 CFR 50.46, Appendix A, GDC 10.
5. 10 CFR 50.46.
7. ~~44~~ 10 CFR 50.46, APPENDIX A, GDC 26.
6. FSAR, SECTION [15].

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Planar Radial Peaking Factors (F_{xy}) (Digital)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

- Using full or part length CEAs to alter the axial power distribution;
- Decreasing CEA insertion by boration, thereby improving the radial power distribution; and
- Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. Limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution. Power distribution is a product of multiple parameters, various combinations of

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BASES

BACKGROUND
(continued)

which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on linear heat rate (LHR) and departure from nucleate boiling (DNB).

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AODs is ~~(1.24)~~ as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) and the core protection calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and the DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR values. The CPCs perform this function by continuously calculating actual values of DNBR and local power density (LPD) for comparison with the respective trip setpoints.

DNBR penalty factors are included in both the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience greater rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

The COLSS indicates continuously to the operator how near the core is to the operating limits and provides an audible

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BASES

BACKGROUND
(continued)

alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP. This threshold is set at 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high LPD or low DNBR trips in the RPS initiate a reactor trip before fuel design limits are exceeded.

The limits on ASI, F_{xy}, and T_g represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE
SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

- ~~Under Review~~
- During a*LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);
 - During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

- c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. ~~11/1~~); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (~~600 RB1~~ Ref. ~~11/1~~).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and the uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_D limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Fuel cladding damage does not occur because of conditions outside the limits of these LCOs for ASI, F_{xy}, and T_D during normal operation. However, fuel cladding damage results if an accident occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased LHR.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

F_{xy} satisfies Criterion 2 of the NRC Policy Statement.

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR.

Limiting of the calculated Planar Radial Peaking Factors (F_{xy}^c) used in the COLSS and CPCs to values equal to or greater than the measured Planar Radial Peaking Factors (F_{xy}^m) ensures that the limits calculated by the COLSS and CPCs remain valid.

APPLICABILITY

Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

- a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate because of the poor signal to noise ratio that they experience at relatively low core power levels; and
- b. As a result of this inaccuracy, the CPCs assume a minimum core power of 20% RTP when generating the LPD and DNBR trip signals. When the core power is below 20% RTP, the core is operating well below its thermal limits, and the resultant CPC calculated LPD and DNBR trips are highly conservative.

ACTIONS

A.1.1 and A.1.2

When the F_{xy}^m values exceed the F_{xy}^c values used in the COLSS and CPCs, nonconservative operating limits and trip setpoints may be calculated. In this case, action must be taken to ensure that the COLSS operating limits and CPC trip setpoints remain valid with respect to the accident

(continued)

BASES

ACTIONS

A.1.1 and A.1.2 (continued)

analysis. The operator can do this by performing the Required Actions A.1.1 and A.1.2. The 6 hour Completion Time provides the time required to calculate the required multipliers and make the necessary adjustments to the CPC addressable constants. During this period the DNBR and LHR setpoints may be slightly nonconservative but DNBR and LHR are still within limits. Therefore, 6 hours is an acceptable Completion Time to perform these actions considering the low probability of an accident occurring during this time period.

A.2

As an alternative to Required Actions A.1.1 and A.1.2, the operator may adjust the affected values of F_{xy}^c used in the COLSS and CPCs to values $\geq F_{xy}^m$. The 6 hour Completion Time provides the time required to calculate the required multipliers and make the necessary adjustments to the CPC addressable constants. During this period the DNBR and LHR setpoints may be slightly nonconservative but DNBR and LHR are still within limits. Therefore, 6 hours is an acceptable Completion Time to perform these actions considering the low probability of an accident occurring during this time period.

A.3

If Required Actions A.1.1 and A.1.2 or A.2 cannot be accomplished within 6 hours, the core power must be reduced. Reduction to 20% RTP or less ensures that the core is operating within the specified thermal limits and places the core in a conservative condition based on the trip setpoints generated by the COLSS and CPC operating limits; these limits are established assuming a minimum core power of 20% RTP. Six hours is a reasonable time to reach 20% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.2.2.1

This periodic Surveillance is for determining, using the Incore Detector System, that F_{xy}^m values are $\leq F_{xy}^c$ values used in the COLSS and CPCs. It ensures that the F_{xy}^c values used remain valid throughout the fuel cycle. A Frequency of 31 EFPD is acceptable because the power distribution changes only slightly with the amount of fuel burnup. Determining the F_{xy}^m values after each fuel loading when THERMAL POWER is $> 40\%$ RTP, but prior to its exceeding 70% RTP, ensures that the core is properly loaded.

REFERENCES

1. FSAR, Section [15].
2. FSAR, Section [6].
3. CE-1 Correlation for DNBR.
4. 10 CFR 50.46, Appendix A, GDC 10.
5. 10 CFR 50.46.
- 7 ~~4~~. 10 CFR 50, APPENDIX A, GDC 26.
6. FSAR, SECTION [15].

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AZIMUTHAL POWER TILT (T_q) (Digital)

SEE NOTE AT
B 3.2.2

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

- Using full or partial ^{sup}length CEAs to alter the axial power distribution;
- Decreasing CEA insertion by boration, thereby improving the radial power distribution; and
- Correcting off optimum conditions, (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

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BASES

BACKGROUND
(continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the linear heat rate (LHR) and the departure from nucleate boiling (DNB).

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions. [] as

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) and the core protection calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and the DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating actual values of DNBR and local power density (LPD) for comparison with the respective trip setpoints.

A DNBR penalty factor is included in the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by the assembly. Fuel assemblies that incur higher than average burnup experience greater magnitude of rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins caused by the lower radial power peaks in the higher burnup batches.

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BASES

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specific limits. The data required for these assessments include measured incore neutron flux data, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicate CEA position. In this case, the CPCs assume a minimum core power of 20% RTP. This threshold is set at 20% RTP because the power range excore neutron flux detection system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on the ASI, F_{xy} , and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of operation and AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

- a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);
- c. During a CEA ejection accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. ~~150~~ ^{L6}); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. ~~150~~ ^{L7}).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 1). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analysis (Ref. 2) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits of these variables ensures that their actual values are within the range used in the accident analyses.

Fuel cladding damage does not occur from conditions outside the limits of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs due to initial conditions outside the limits of these LCOs. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

T_a satisfies Criterion 2 of the NRC Policy Statement.

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR.

The limitations on the T_a are provided to ensure that design operating margins are maintained. $T_a > 0.10$ is not expected. If it occurs, the actions to be taken ensure that operation is restricted to only those conditions required to identify the cause of the tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

APPLICABILITY

Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

- a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate due to the poor signal to noise ratio that they experience at relatively low core power levels.
- b. As a result of this inaccuracy, the CPCs assume a minimum core power of 20% RTP when generating LPD and DNBR trip signals. When the core power is below this level, the core is operating well below its thermal limits and the resultant CPC calculated LPD and DNBR trips are highly conservative.

ACTIONS

A.1 and A.2

If the measured T_a is greater than the T_a allowance used in the CPCs but ≤ 0.10 , nonconservative trip setpoints may be calculated. Required Action A.1 restores T_a to within its

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

specified limits by repositioning the CEAs, and the reactor may return to normal operation. A Completion Time of 2 hours is sufficient time to allow the operator to reposition the CEAs because significant radial xenon redistribution does not occur within this time.

If the T_a cannot be restored within 2 hours, the T_a allowance in the CPCs must be adjusted, per Required Action A.2, to be equal to or greater than the measured value of T_a to ensure that the design safety margins are maintained.

B.1, B.2, and B.3

Required Actions B.1, B.2, and B.3 are modified by a Note that requires all subsequent actions be performed if power reduction commences prior to restoring $T_a \leq 0.10$. This requirement ensures that corrective action is taken before unrestricted power operation resumes.

If the measured $T_a > 0.10$, THERMAL POWER is reduced to $\leq 50\%$ RTP within 4 hours. The 4 hours allows enough time to take action to restore T_a prior to reducing power and limits the probability of operation with a power distribution out of limits. Such actions include performing SR 3.2.3.2, which provides a value of T_a that can be used in subsequent actions.

Also in the case of a tilt generated by a CEA misalignment, the 4 hours allows recovery of the CEA misalignment, because a measured $T_a > 0.10$ is not expected. If it occurs, continued operation of the reactor may be necessary to discover the cause of the tilt. Operation then is restricted to only those conditions required to identify the cause of the tilt. It is necessary to explicitly account for power asymmetries because the radial power peaking factors used in the core power distribution calculation are based on an untilted power distribution.

If the measured T_a is not restored to within its specified limits, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation, which

(continued)

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

results in increased linear heat generation rates when the xenon redistributes. If the measured T_a cannot be restored to within its limit within 4 hours, reactor power must be reduced. Reducing THERMAL POWER to < 50% RTP within 4 hours provides an acceptable level of protection from increased power peaking due to potential xenon redistribution while maintaining a power level sufficiently high enough to allow the tilt to be analyzed.

The Linear Power Level-High trip setpoints are reduced to ≤ 55% RTP to ensure that the assumptions of the accident analysis regarding power peaking are maintained. After power has been reduced to ≤ 50% RTP, the rate and magnitude of changes in the core flux are greatly reduced. Therefore, 16 hours is an acceptable time period to allow for reduction of the Linear Power Level-High trip setpoints. Required Action B.2. The 16 hour Completion Time allowed to reduce the Linear Power Level-High trip setpoints is required to perform the actions necessary to reset the trip setpoints.

THERMAL POWER is restricted to 50% RTP until the measured T_a is restored to within its specified limit by correcting the out of limit condition. This action prevents the operator from increasing THERMAL POWER above the conservative limit when a significant T_a has existed, but allows the unit to continue operation for diagnostic purposes.

The Completion Time of Required Action B.3 is modified by a Note governing subsequent power increases. After a THERMAL POWER increase following restoration of T_a, operation may proceed provided the measured T_a is determined to remain within its specified limit at the increased THERMAL POWER level.

The provision to allow discontinuation of the surveillance after verifying that T_a ≤ 0.10 is within its specified limit at least once per hour for 12 hours or until T_a is verified to be within its specified limit at a THERMAL POWER ≥ 95% RTP provides an acceptable exit from this Action after the measured T_a has been returned to an acceptable value.

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BASES

ACTIONS (continued)

C.1

If the measured T_a cannot be restored or determined within its specified limit, core power must be reduced. Reduction of core power to < 20% RTP ensures that the core is operating within its thermal limits and places the core in a conservative condition based on the trip setpoints generated by the CPCs, which assume a minimum core power of 20% RTP. Six hours is a reasonable time to reach 20% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

Continuous monitoring of the measured T_a by the incore nuclear detectors is provided by the COLSS. A COLSS alarm is annunciated in the event that the measured T_a exceeds the value used in the CPCs.

With the COLSS out of service, the operator must calculate T_a and verify that it is within its specified limits. The 12 hour Frequency is sufficient to identify slowly developing T_a 's before they exceed the limits of this LCO. Also, the 12 hour Frequency prevents significant xenon redistribution.

SR 3.2.3.2

Verification that the COLSS T_a alarm actuates at a value less than the value used in the CPCs ensures that the operator is alerted if T_a approaches its operating limit. The 31 day Frequency for performance of this SR is consistent with the historical testing frequency of reactor protection and monitoring systems. The Surveillance Frequency for testing protection systems was extended to 92 days by CEN 327. Monitoring systems were not addressed in CEN 327, therefore this Frequency remains at 31 days.

SR 3.2.3.3

Independent confirmation of the validity of the COLSS calculated T_a ensures that the COLSS accurately identifies T_a 's.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.3.3 (continued)

The 31 day Frequency for performance of this SR is consistent with the historical testing frequency of reactor protection and monitoring systems. The Surveillance Frequency for testing protection systems was extended to 92 days by CEN 327. Monitoring systems were not addressed in CEN 327, therefore this Frequency remains at 31 days.

REFERENCES

1. FSAR, Section [15].
 2. FSAR, Section [6].
 3. CE-1 Correlation for DNBR.
 4. 10 CFR 50.46, Appendix A, GDC 10.
 5. 10 CFR 50.46.
 6. 10 CFR 50, Appendix A, GDC 26.
 7. FSAR, SECTION [15].
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Departure from Nucleate Boiling Ratio (DNBR) ~~(Digital)~~

BASES

SEE NOTE AT B 3.2.2 BACKGROUND

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial value assumed in the accident analyses. Specifically, operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

- Using full or part length CEAs to alter the axial power distribution;
- Decreasing CEA insertion by boration, thereby improving the radial power distribution; and
- Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analysis (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling axial power distribution.

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BASES

BACKGROUND
(continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the linear heat rate (LHR) and the departure from nucleate boiling (DNB).

Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and ADOs is ~~1.4~~ as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bows and grid spacers and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limits Supervisory System (COLSS) and the core protection calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating an actual value of DNBR and LPD for comparison with the respective trip setpoints.

A DNBR penalty factor is included in both the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience a greater magnitude of rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty that is applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

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BASES

BACKGROUND
(continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm when an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded during AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to the exceeding of fuel design limits.

The limits on ASI, F_{xy} , and T_a represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE
SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

- a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);
- b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4):

- c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 4). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_g limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analyses (Ref. 1).

Fuel cladding damage does not occur from conditions outside the limits of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

DNBR satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR.

With the COLSS in service and one or both of the Control Element Assembly Calculators (CEACs) OPERABLE, the DNBR will be maintained by ensuring that the core power calculated by the COLSS is equal to or less than the permissible core power operating limit calculated by the COLSS. In the event that the COLSS is in service but neither of the two CEACs is OPERABLE, the DNBR is maintained by ensuring that the core power calculated by the COLSS is equal to or less than a reduced value of the permissible core power operating limit calculated by the COLSS. In this condition, the calculated operating limit must be reduced by 13.0% RTP.

In instances for which the COLSS is out of service and either one or both of the CEACs are OPERABLE, the DNBR is maintained by operating within the acceptable region specified in the COLR as shown in Figure 3.2.4-1, in the COLR, and using any OPERABLE CPC channel. Alternatively, when the COLSS is out of service and neither of the two CEACs is OPERABLE, the DNBR is maintained by operating within the acceptable region specified in the COLR for this condition as shown in Figure 3.2.4-2, in the COLR, and using any OPERABLE CPC channel.

With the COLSS out of service, the limitation on DNBR as a function of the ASI represents a conservative envelope of operating conditions consistent with the analysis assumptions that have been analytically demonstrated adequate to maintain an acceptable minimum DNBR for all AOOs. Of these, the postulated loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit ensures that an acceptable minimum DNBR is maintained in the event of a loss of flow transient.

APPLICABILITY

Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only

(continued)

BASES

APPLICABILITY (continued)

applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

- a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate due to the poor signal to noise ratio that they experience at relatively low core power levels.
- b. As a result of this inaccuracy, the CPCs assume a minimum core power of 20% RTP when generating the local power density (LPD) and DNBR trip signals. When the core power is below this level, the core is operating well below the thermal limits and the resultant CPC calculated LPD and DNBR trips are highly conservative.

ACTIONS

A.1

Operating at or above the minimum required value of the DNBR ensures that an acceptable minimum DNBR is maintained in the event of a postulated loss of flow transient. If the core power as calculated by the COLSS exceeds the core power limit calculated by the COLSS based on the DNBR, fuel design limits may not be maintained following a loss of flow, and prompt action must be taken to restore the DNBR above its minimum Allowable Value. With the COLSS in service, 1 hour is a reasonable time for the operator to initiate corrective actions to restore the DNBR above its specified limit, because of the low probability of a severe transient occurring in this relatively short time.

B.1

If the COLSS is not available the OPERABLE DNBR channels are monitored to ensure that the DNBR is not exceeded. Maintaining the DNBR within this specified range ensures that no postulated accident results in consequences more severe than those described in the FSAR, Chapter 15. A 4 hour Frequency is allowed to restore the DNBR limit to within the region of acceptable operation. This Frequency is reasonable because the COLSS allows the plant to operate with less DNBR margin (closer to the DNBR limit) than when monitoring with the CPCs.

(continued)

BASES

ACTIONS

B.1 (continued)

W. J. Taylor
2 hours

Also, when operating with the COLSS out of service there is a possibility of a slow undetectable transient that degrades the DNBR slowly over the 4 hour period and is then followed by an anticipated operational occurrence or an accident. Therefore, the CPC calculated values of DNBR are monitored every 15 minutes when the COLSS is out of service. Also, a maximum allowable change in the CPC calculated DNBR ensures that further degradation requires the operators to take immediate action to reduce reactor power to comply with the technical specifications. Implementation of this requirement ensures that potential reductions in core thermal margin are quickly detected and, if necessary, cause a decrease in reactor power and subsequent compliance with the existing COLSS out of service Technical Specification limits.

Four hours is allowed for restoring the DNBR to within limits if the COLSS is not restored to OPERABLE status. This duration is reasonable because the Frequency of the CPC determination of DNBR has been increased, and, with the operation maintained steady, the likelihood of exceeding the DNBR limit during the additional 2 hours is not increased. Also, the likelihood of induced reactor transients from an early power reduction is decreased.

C.1

If the DNBR cannot be restored or determined within the allowed times of Conditions A and B, core power must be reduced. Reduction of core power to < 20% RTP ensures that the core is operating within its thermal limits and places the core in a conservative condition based on trip setpoints generated by the CPCs, which assume a minimum core power of 20% RTP.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 20% RTP from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTSSR 3.2.4.1

With the COLSS out of service, the operator must monitor the DNBR as indicated on any of the OPERABLE DNBR channels of the CPCs to verify that the DNBR is within the specified limits, shown in either Figure 3.2.4-1 or 3.2.4-2 of the COLR, as applicable. A 2 hour Frequency is adequate to allow the operator to identify trends in conditions that would result in an approach to the DNBR limit.

This SR is modified by a Note that states that the SR is only applicable when the COLSS is out of service. Continuous monitoring of the DNBR is provided by the COLSS, which calculates core power and core power operating limits based on the DNBR and continuously displays these limits to the operator. A COLSS margin alarm is annunciated in the event that the THERMAL POWER exceeds the core power operating limit based on the DNBR.

SR 3.2.4.2

Verification that the COLSS margin alarm actuates at a power level equal to or less than the core power operating limit, as calculated by the COLSS, based on the DNBR, ensures that the operator is alerted when operating conditions approach the DNBR operating limit. The 31 day Frequency for performance of this SR is consistent with the historical testing frequency of reactor protection and monitoring systems. The surveillance frequency for testing protection systems was extended to 92 days by CEN 327. Monitoring systems were not addressed in CEN 327; therefore, this Frequency remains at 31 days.

REFERENCES

1. FSAR, Chapter [15].
2. FSAR, Chapter [6].
3. C-E 1 Correlation for DNBR.
4. 10 CFR 50, Appendix A, GDC 10.
5. 10 CFR 50.46.

(continued)

BASES

REFERENCES
(continued)

6. FSAR, Section [1.5].
 7. 10 CFR 50, Appendix A, GDC 26.
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 AXIAL SHAPE INDEX (ASI) ~~(Digital)~~

BASES

See note at B.3.2.2

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

- Using full or part ^{step} length CEAs to alter the axial power distribution;
- Decreasing CEA insertion by boration, thereby improving the radial power distribution; and
- Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Minimizing power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BASES

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the linear heat rate (LHR) and the departure from nucleate boiling (DNB).

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and ADOs is ~~[1.15]~~ as calculated by the CE-1 Correlation (Ref. 3), and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) or the core protection calculators (CPCs). The COLSS and CPCs monitor the core power distribution and are capable of verifying that the LHR and DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating actual values of DNBR and local power density (LPD) for comparison with the respective trip setpoints.

A DNBR penalty factor is included in both the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience greater rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty that is applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPC is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

(continued)

BASES

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to the exceeding of fuel design limits.

The limits on ASI, F_{xy} , and T_g represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

- a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);
- b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

- c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6);
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations among measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_o limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analysis.

Fuel cladding damage does not occur from conditions outside these LCOs during normal operation. However, fuel cladding damage results when an accident occurs due to initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

The ASI satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR.

The limitation on ASI ensures that the actual ASI value is maintained within the range of values used in the accident analysis. The ASI limits ensure that with T_g at its maximum upper limit, the DNBR does not drop below the DNBR Safety Limit for AOOs.

APPLICABILITY Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

- a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate due to the poor signal to noise ratio that they experience at relatively low core power levels.
- b. As a result of this inaccuracy, the CPCs assume a minimum core power of 20% RTP when generating the LPD and DNBR trip signals. When the core power is below this level, the core is operating well below the thermal limits and the resultant CPC calculated LPD and DNBR trips are strongly conservative.

ACTIONS

A.1

The ASI limits specified in the COLR ensure that the LOCA and loss of flow accident criteria assumed in the accident analyses remain valid. If the ASI exceeds its limit, a Completion Time of 2 hours is allowed to restore the ASI to within its specified limit. This duration gives the operator sufficient time to reposition the regulating or part-length CEAs to reduce the axial power imbalance. The magnitude of any potential xenon oscillation is significantly reduced if the condition is not allowed to persist for more than 2 hours.

(continued)

BASES

ACTIONS (continued)

B.1

If the ASI is not restored to within its specified limits within the required Completion Time, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration induces an axial xenon oscillation, and results in increased linear heat generation rates when the xenon redistributes. Reducing thermal power to $\leq 20\%$ RTP reduces the maximum LHR to a value that does not exceed the fuel design limits if a design basis event occurs. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

The ASI can be monitored by both the incore (COLSS) and excore (CPC) neutron detector systems. The COLSS provides the operator with an alarm if an ASI limit is approached.

Verification of the ASI every 12 hours ensures that the operator is aware of changes in the ASI as they develop. A 12 hour Frequency for this Surveillance is acceptable because the mechanisms that affect the ASI, such as xenon redistribution or CEA drive mechanism malfunctions, cause slow changes in the ASI, which can be discovered before the limits are exceeded.

REFERENCES

1. FSAR, Chapter [15].
2. FSAR, Chapter [6].
3. CE-1 Correlation for DNBR.
4. 10 CFR 50, Appendix A, GDC 10.
5. 10 CFR 50.46.

(continued)

BASES

REFERENCES
(continued)

6. FSAR, Section [15].
 7. 10 CFR 50, Appendix A, GDC 26.
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ATTACHMENT 3

9.0 AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

9.1.1 NEW FUEL STORAGE

9.1.1.1 Design Bases

The following design bases are imposed on the storage of new fuel assemblies:

- B
- A. Accidental criticality shall be prevented for the most reactive arrangement of new fuel stored. For normal operation and postulated accident conditions identified in Section 9.1.1.3.1.1 A, C and D, K_{eff} shall be maintained less than 0.95. For the postulated accident condition identified in Section 9.1.1.3.1.1^B, K_{eff} shall be maintained less than 0.98.
 - B. All requirements of Regulatory Guide 1.13 are met excluding those regarding the spent fuel pool water supply, since new fuel storage is dry. The new fuel storage area is designed to ensure that any light load, when handled over the fuel racks, will not exceed the design impact energy of the rack if the load is postulated to fall from its operating height. In addition, all heavy loads are prevented from travel over the new fuel racks by the use of mechanical and electrical interlocks on the cask handling hoist. The new fuel handling hoist incorporates load limiting devices to preclude fuel damage during handling.
 - C. The storage racks and facilities are (See Section 9.1.1.3.3) qualified as Seismic Category I_o per Regulatory Guide 1.29.
 - D. Storage is typically provided for 121 fuel assemblies. This capacity, which represents 50% of the fuel assemblies in the core, envelops any reload batch size that would occur for refueling cycle lengths up to and including 24 months.
 - E. The fuel handling equipment located in the new fuel storage area meets the requirements of ANS 57.1. The new fuel racks meet the requirements of ANS 57.3.
 - F. The New Fuel Storage Racks are designed to meet the requirements of SRP 3.8.4 Appendix D which addresses appropriate combinations of seismic and dropped loads with allowable stress/deformation limits.

- C. Providing positive hoist travel limits and interlocks to ensure proper equipment operation and sequencing.
- D. Limiting the insertion loads when installing fuel into the new fuel racks by load measuring devices or hoist underload interlocks.
- E. Designing the new fuel racks for Safe Shutdown Earthquake (SSE) conditions and dropped fuel assembly conditions, considered separately.
- F. Designing the new fuel handling hoist to preclude the hoist, or any part thereof, from falling into the new fuel handling area.
- G. Providing a drain line at the bottom of the storage cavity to direct any fluid to the floor drain sump. The drain incorporates a non-return check valve.
- H. Designing the building with no water source in the immediate area of the new fuel storage racks. The fixed fire fighting water supply standpipes are seismically designed (SSE).
- I. Restricting the lifting capacity of the new fuel handling hoist that is used to remove new fuel assemblies from the new fuel rack by either adjusting the motor stall torque or using load limiting devices. (See Paragraphs 9.1.1.3.1.1.D and 9.1.4.2.1.7.B). Therefore excessive uplifting force cannot be applied.
- J. Locating the new fuel storage racks at the opposite end of the fuel building from the spent fuel pool to eliminate the possibility of moving heavy loads near the new fuel storage area. Movement of heavy loads over the fuel racks is restricted by the use of electrical interlocks on the cask handling hoist.
- K. Locating the new fuel storage racks in a concrete vault at the opposite end of the fuel area of the nuclear annex from the spent fuel pool area to preclude passage of the ~~spent fuel shipping~~ cask ~~overhead~~ hoist over the new fuel racks during the handling operations associated with spent fuel inspection, handling, and shipping. This location minimizes the number of systems or structures located in the vicinity of the new fuel storage facility. All systems or structures in the vicinity will be designated as Seismic Category II to preclude their failure and entry into the new fuel storage area.

9.1.1.3.3 Seismic Classification

The New Fuel Storage Racks, Storage Vault, and the Rack Restraint System are qualified as Seismic Category I. The seismic category of other building components associated with handling fuel assemblies is noted in Table 3.2-1. Those items in the immediate vicinity of the new fuel storage area that are not qualified as Seismic Category I are designed such that their failure will not result in damage to the fuel racks or fuel (See Section 9.1.1.3.1.1.J).

9.1.1.3.4 Storage Capacity

Storage is typically provided for a total of 121 new fuel assemblies in two 50% density 11x11 racks.

9.1.2 SPENT FUEL STORAGE

9.1.2.1 Design Bases

The following design bases are imposed on the storage of fuel within the spent fuel pool:

- A. Accidental criticality shall be prevented for the most reactive arrangement of stored spent fuel by avoiding a K_{eff} greater than 0.95. This design basis shall be met under any normal operation and postulated accident conditions identified in Section 9.1.2.3.1.1. 1.29, 1.115 and 1.117
- B. The requirements of Regulatory Guide 1.13 shall be met. The spent fuel pool area is designed to prevent a loss of water in the fuel pool from uncovering the fuel, prevent heavy loads from traversing over the fuel racks when the racks contain fuel assemblies, withstand the impact of a fuel assembly or a handling tool or a combination of both falling from the maximum handling elevation, incorporate components meeting the seismic classification designated in Table 3.2-1, and incorporate water level and radiation monitoring instrumentation.
- C. The storage racks and facilities shall be Seismic Category I.
- D. Storage shall be provided for up to 907 spent fuel assemblies. All components within the area of the fuel racks meet the requirements of Table 3.2-1 to preclude rack damage.
- E. The racks shall not be anchored to the pool floor or wall. Clearances shall be allowed for rack tipping but the rack design and loading shall preclude rack overturning.

9.1.2.3 Safety Evaluation

The spent fuel pool storage rack design and location, discussed in Section 9.1.2.2, provides assurance that design bases of Section 9.1.2.1 are met as noted in the following sections.

9.1.2.3.1 Criticality Safety**9.1.2.3.1.1 Postulated Accidents**

The following postulated accidents are considered in the design of the spent fuel pool storage racks:

- A. A load dropped on the loaded fuel racks whose impact energy, if dropped from the operating elevation, will not exceed the impact energy of the postulated dropped fuel handling tool or the combination of the dropped fuel handling tool, fuel assembly, and any other fuel handling component supported by the hoist cabling.
- B. Tensile load on the rack of 5000 pounds (limited by adjustment of the motor stall torque or load-limiting device of the crane used to load fuel into the racks). (See Section 9.1.4.2.1.3)
- C. A fuel assembly accidentally located in either a blocked off fuel storage cavity or adjacent to the outside of the fuel rack.

Although the above accident conditions have been postulated, the fuel handling equipment, fuel racks, and the building arrangement are designed to minimize the possibility of these accidents or the effects resulting from these accidents by:

- A. Providing positive mechanical travel hoist limits and interlocks to ensure proper equipment operation and sequence.
- B. Limiting the handling of loads when installing fuel into or removing fuel from the fuel rack.
- C. Designing the fuel racks for (1) SSE conditions and (2) a dropped fuel assembly handling tool or the combination of the dropped fuel handling tool, fuel assembly, and any other component supported by the hoist cable (conditions (1) and (2) considered separately).
- D. Designing the fuel handling machine as a Seismic Category II to preclude the fuel handling machine, or any part thereof, from falling into the spent fuel pool.

- H. Providing mechanical and electrical interlocks on the nuclear annex overhead ~~cranes~~ to preclude movement of fuel shipping containers or casks and other heavy loads from being transported over the spent fuel pool. (See Section 9.1.4.2.1.7)
- E. Meeting regulatory positions C.1 and C.2 of Regulatory Guide 1.29 and regulatory positions C.1 and C.6 of Regulatory Guide 1.13, as these positions relate to the ability of the components to withstand the effects of earthquakes.

Examples of compliance are demonstrated by the assignment of the various Seismic Categories to the building structures, fuel handling equipment, and other components as noted in Table 3.2-1 and the design of the equipment and components meeting these requirements. Fuel handling equipment that moves over the reactor core and spent fuel racks is also provided with seismic restraints to ensure that the components do not become disengaged from their operating rails and fall into the pool during a seismic event.

- F. Meeting regulatory positions C.1, C.2 and C.3 of Regulatory Guide 1.13; ANS 57.1/ANSI-N028, ANS 57.2/ANSI-N210, NUREG-0554, and NUREG-0612 as they relate to radioactive release as a result of fuel damage.

Examples of compliance are demonstrated by the design of the fuel building which precludes movement of the spent fuel cask handling hoist over the new and spent fuel storage racks when they contain fuel assemblies, designation of load paths for all heavy lifts, limiting the weight and lift height of any load that is moved over the fuel racks such that its impact energy, if dropped, will not exceed the design impact energy of the fuel racks (See Section 9.1.2.3.1.1.G) or fuel pool, and ensuring that the lift height of the spent fuel shipping cask does not exceed 30 feet which limits the cask from being raised above the operating floor elevation.

- G. Permitting no load to be carried over the loaded fuel racks whose impact energy, if dropped from the operating elevation, will exceed the impact energy of the postulated dropped fuel handling tool, fuel assembly, and any other handling component supported by the hoist cabling when lifting fuel assemblies. The Technical Specification incorporates the requirement that the impact energy of all loads carried over the loaded fuel racks will not exceed this condition.

9.1.2.3.1.2 Criticality Safety Assumptions

The following assumptions are made in evaluating criticality safety:

- A. No control element assemblies (CEAs) are assumed to be present in the fuel assemblies.

conditions (e.g., variations in fuel rack pitch, rack steel thickness, spent fuel pool water temperature) and calculational uncertainties. Including all uncertainties, the maximum K_{eff} for Region I is less than 0.95.

For Region II, K_{eff} is calculated for various combinations of fuel enrichments and fuel burnups, with allowance for uncertainties due to deviations from nominal conditions and calculational uncertainties. The initial enrichments range up to and including 5 wt.% U-235. These results conservatively establish the minimum cumulative burnup as a function of initial enrichment for Region II fuel necessary to maintain K_{eff} less than 0.95. (For conservatism, the minimum cumulative burnup represents 0.85 of the actual fuel burnup.)

The three-dimensional Monte Carlo Computer code KENO-IV (Reference 2) is used to calculate K_{eff} for the postulated accident condition of a dropped fuel assembly in Region II. The dropped fuel assembly is conservatively assumed to be a fresh fuel assembly with 5.0 wt.% U-235 initial enrichment. The assumed boron concentrations are significantly conservative (less than one half of the minimum boron concentration required by the Technical Specifications) with respect to the actual boron concentrations that could occur during the postulated dropped fuel assembly accident. With these assumed boron concentrations, the maximum K_{eff} (including uncertainties) for the postulated dropped fuel assembly accident condition is substantially less than 0.95.

Thus, for normal operation and postulated accident conditions, K_{eff} is shown to be less than 0.95. The K_{eff} values are substantially below the limiting values allowed by ANS/ANSI 51.1-1983 and provide adequate margin for calculation uncertainty.

The spent fuel storage area is protected from the effects of missiles or natural phenomena by a Seismic Category I structure, as discussed in Section 3.5.

9.1.2.3.2 Compliance with Regulatory Guide 1.13

The spent fuel storage facility ^{conforms with the guidelines} ~~complies with the intent~~ of Regulatory Guide 1.13.

9.1.2.3.3 Seismic Classification

The spent fuel storage racks, the spent fuel pool concrete structure, the spent fuel rack support system, and the pool liner are Seismic Category I. Refer to Table 3.2-1 for a tabulation of the designated seismic categories for the fuel handling and nuclear annex components related to fuel handling.

9.1.2.3.4 Storage Capacity

Storage is provided for up to 907 spent fuel assemblies. This provides storage for approximately 10 years of unit operation.

9.1.2.3.5 Fuel Assembly Cooling

The spent fuel pool storage racks are designed to prevent extensive bulk boiling in the racks as well as maintain fuel cladding temperatures well below 650°F for the following collective conditions:

- A. Natural convection water circulation within the spent fuel pool,
- B. Maximum pool water temperature of 150°F at the fuel rack inlet flow passages, and
- C. Maximum fuel pool heat load as described in Section 9.1.3.

9.1.2.3.6 Compliance with ANS 57.2

The design of the Spent Fuel Storage facility ^{conforms with the guidelines} ~~meets the intent~~ of paragraph 5.4 of ANS 57.2. As an example, the facility incorporates monitoring systems to verify pool water temperature to insure adequate fuel assembly cooling, radiation detectors to determine if radiation levels exceed predetermined setpoints and alarms to notify plant personnel of abnormal conditions.

The features include:

- A. A radiation monitor with audible alarm on the spent fuel handling machine adjacent to the operator control console.
- B. Radiation monitors, including a continuous air monitor, within the spent fuel pool area. At least one monitor indicates and alarms in the control room.
- C. Uninterruptible communications by the use of sound powered phones or a separate communication system.
- D. Redundant alarm and actuation system for the pool ventilation system during those periods it is not in use.
- E. Ventilation sampling provisions.

To facilitate use, all monitoring systems are capable of being calibrated.

REFERENCES FOR SECTION 9.1

1. W. A. Rhoades and R. L. Childs, "An Updated Version of the DOT-4 One- and Two-Dimensional Neutron/Photon Transport Code", ORNL-5851, April 1982.
2. L. M. Petrie and N. F. Cross, "KENO-IV, An Improved Monte Carlo Criticality Program", ORNL-4938, November 1975.

9.1.1.3.1.2

Criticality Safety Assumptions

The following assumptions are made in evaluating criticality safety:

- A. Under postulated conditions of complete ^{a finite} flooding by unborated water, the storage array is treated as ~~an infinite~~ array of assemblies having an infinite fuel length.
- B. Under postulated conditions of envelopment by aqueous foam or mist, a range of foam or mist densities is examined to ensure that the maximum reactivity of the array is established. The foam or mist is assumed to be pure water.
- C. The poisoning effects of rack structure are neglected. Prior calculations have shown this to be a conservative assumption, where the degree of conservatism depends on the exact rack structure design. It is also assumed that no supplemental fixed poisons are utilized in the storage array.
- D. A concrete storage cavity is utilized for new fuel storage. Two 11x11 rack modules are located in the cavity with cell blockers installed in alternate cells to limit new fuel storage to 121 fuel assemblies.

The criticality analyses for the new fuel racks assume a close-fitting, 2-foot thick concrete reflector on all six sides of the new fuel rack array. In actuality, the concrete walls surrounding the new fuel racks are separated from the racks by several inches, with the floor and material above the fuel also several inches away from the racks. A close fitting, thick concrete wall provides better neutron reflection than both the reflector consisting of a concrete wall separated from the array by several inches and the reflector consisting of the actual materials above the active fuel. Therefore, the configuration assumed for the criticality analyses is conservative with respect to the actual configuration of the new fuel rack array.

- E. The rack is assumed to be filled to design capacity with fuel assemblies.
- F. No burnable poison shims or other supplemental neutron poisons (e.g., CEAs) are assumed to be present in the fuel assemblies.

9.1.4 FUEL HANDLING SYSTEM**9.1.4.1 Design Bases****9.1.4.1.1 System**

The fuel handling system is designed for the handling and storage of fuel assemblies and control element assemblies (CEAs). Associated with the fuel handling system is the equipment used for assembly, disassembly, and storage of the reactor closure head and internals. As appropriate, the fuel handling equipment includes interlocks, travel-limiting features, and other protective devices to minimize the possibility of mishandling or equipment malfunction that could result in inadvertent damage to a fuel assembly and potential fission product release.

The refueling water provides the coolant medium during spent fuel transfer. The spent fuel pool is provided with a cooling and cleanup system.

All spent fuel transfer and storage operations are designed to be conducted underwater to ensure adequate shielding during refueling and to permit visual control of the operation at all times.

9.1.4.1.2 Fuel Handling Equipment

The principal design criteria for the fuel and CEA handling equipment (refueling machine, fuel transfer equipment, spent fuel handling machine, CEA change platform, new fuel elevator, and CEA elevator) are as follows:

- A. For non-seismic operating conditions, the bridges, trolleys, hoist units, hoisting cable, grapples, and hooks conform to the requirements of Crane Manufacturing Association of America Specification #70.
- B. For seismic design, the combined dead loads, live loads, and seismic loads do not cause any portion of the equipment to disengage from its supports and fall into the pool.
- C. Grapples and mechanical latches which carry fuel assemblies or CEAs are mechanically interlocked against inadvertent opening.
- D. All loose components are either removed from equipment in the reactor building and the fuel area of the nuclear annex
OR ~~not~~ are seismically restrained during equipment operation. Fuel handling procedures require that lanyards be used for loose components that are brought onto the equipment for a particular short term application such as repair tooling.

All permanently installed components are secured with locking devices or restraints to prevent them from becoming loose and falling into the refueling pool or the spent fuel pool.

- E. A positive mechanical stop is provided to prevent the fuel from being lifted above the minimum safe water cover depth and it will not cause damage or distortion to the fuel or the refueling machine when engaged at full operating hoist speed.
- F. The fuel hoists are provided with load-measuring devices and interlocks to interrupt hoisting if the load increases above the overload setpoint and to interrupt lowering if the load decreases below the underload setpoint.
- G. In the event of loss of power, the equipment and its load remain in a safe condition.
- H. Equipment located within the reactor building during plant operation is capable of withstanding, without damage, the internal building test pressure.
- I. Electrical interlocks are provided to ensure the reliability of system components, to simplify the performance of sequential operations, and to limit travel and loads such that design conditions will not be exceeded. In no case will they be utilized to prevent inadvertent criticality or the reduction of the minimum water coverage for personnel protection. No single interlock failure will result in a condition which will allow equipment malfunction, damage to the fuel, or the reduction of shielding water coverage. Where these results are considered possible, redundant switches, mechanical restraints, and physical barriers are employed as well as limiting the hoist stall torque and loading capability to values below those which would result in damage to the fuel.

9.1.4.1.3

~~Fuel Building~~ ^{Nuclear Annex}

~~Overhead Hoists and Containment~~ ^{Reactor Building}
Polar Crane

The fuel related hoists in the nuclear annex, i.e., the cask handling hoist and the new fuel handling hoist, are used to handle equipment, tools, and fuel assemblies from the receipt of the new fuel containers to the shipment of the spent fuel cask. A description of these hoist is contained in Section 9.1.4.2.1.7. The reactor building polar crane is used to handle the reactor vessel head, reactor vessel internals, and other equipment located within the reactor building. A description of the reactor polar crane is contained in Section 9.1.4.2.1.8.

BUILDING

well as a cooling medium for removal of decay heat. Boric acid is added to the spent fuel pool (SFP) water in the quantity required to assure subcritical conditions.

The major components of the system are the refueling machine (Figure 9.1-4), the CEA change platform (Figure 9.1-5), the fuel transfer system (Figures 9.1-6a and 9.1-6b), the spent fuel handling machine, the CEA elevator (Figure 9.1-7), the new fuel elevator (Figure 9.1-8), nuclear annex overhead cranes and the reactor building polar crane. The refueling machine moves fuel assemblies into and out of the core and between the core and the transfer system. The CEA change platform is used to move the CEAs within the upper guide structure (UGS) or between the UGS and the CEA elevator. The CEA elevator is used to assemble and introduce new CEAs into the refueling pool and to hold the spent CEAs while they are being disassembled for disposal. The fuel transfer system moves the fuel between the reactor building and the reactor annex through the transfer tube. The spent fuel handling machine transports fuel between the transfer system, the spent fuel storage racks, the new fuel elevator, and the spent fuel shipping cask. The new fuel elevator is used to introduce new fuel into the spent fuel pool so that it can be moved to the spent fuel storage racks or the transfer system by the spent fuel handling machine. The cask handling hoist is used to transfer the spent fuel shipping cask between the cask laydown area, cask washdown area, and the fuel building loading bay. The fuel handling hoist is used for handling the new fuel container and new fuel assemblies during transfer from the new fuel shipping container to the new fuel elevator, the new fuel storage racks, or the new fuel inspection station.

The fuel handling hoist and the cask handling hoist are mounted on the same trolley assembly.

An intermediate fuel storage rack is located within the refueling cavity for the temporary storage of fuel assemblies if required, during the refueling operation. Any stored assemblies are removed from the rack prior to reinstallation of the reactor vessel internals. A storage rack for holding the ICI/CEA transport container during ICI and CEA disposal operations is also located within the refueling cavity. Since the transport container is essentially the same size as a fuel assembly for compatibility with the transfer system, the rack is designed to contain a fuel assembly in the event a fuel assembly is inadvertently placed within it.

Special tools and lift rigs are also used for disassembly of reactor components.

Major tools and servicing equipment utilized for refueling are listed in Table 9.1-1.

information to allow the operator to deduce that an interlock has malfunctioned. The interlock functions required by ANS 57.1 are included in the fuel handling equipment design.

The fuel and CEA handling machines do not fully fall within the framework of an overhead or gantry crane as described in OSHA Subpart N, Materials Handling and Storage, of 29 CFR 1910, Section 1910.179. However, this document has been used for guidance. More than 95% of the fuel handling machine does conform to the OSHA regulations. Both machines have additional features to protect the safety of the operator and facility, and the features are a part of appropriate operational procedures.

The two cavity transfer system fuel carrier, the reactor building intermediate fuel storage rack and the transport container storage rack are designed to meet the same criticality considerations as the spent fuel storage racks (Section 9.1.2).

The design of the Fuel Handling System limits the impact energy of postulated dropped loads on the new fuel storage racks, the spent fuel storage racks, the fuel transfer system fuel carrier, and the spent fuel pool. The spent fuel shipping cask is prevented from travelling over the new fuel storage racks and the spent fuel storage racks by mechanical stops and electrical interlocks. The defined load path prevents the shipping cask from traveling within 15 feet of the edge of the spent fuel pool and ~~does not~~ require that the shipping cask be lifted above the operating floor elevation. ^{Not} THIS RESTRICTION ON THE LIFT HEIGHT PRECLUDES PASSAGE OF THE SHIPPING CASK OVER THE NEW FUEL RACKS.

All loads that may be handled over the new fuel storage racks, the spent fuel storage racks, the spent fuel pool and the fuel transfer system fuel carrier are limited in weight and lift height such that, if they fall, the resultant impact energy will not exceed the design impact energy of the fuel storage racks and the spent fuel pool. The design impact energy shall be equal to the postulated drop of a fuel assembly, its handling tool or a combination of both the tool and the fuel assembly, and any other fuel handling component attached to the hoisting cable during fuel assembly handling, from their maximum lifted elevation above the fuel racks during normal handling. The elevation to which the fuel assembly may be lifted is limited by interlocks on the spent fuel handling machine and the new fuel handling hoist (Section 9.1.4.2.1) and the design of the handling tools. The weight that may be lifted is limited by load interlocks and/or hoist motor stall torque.

9.1.4.2.1.1 Refueling Machine

The following identifies and describes the functions of the interlocks which will be contained in the refueling machine.

COL APPLICANT

The Owner-Operator will prepare and implement administrative controls to restrict operation of the fuel transfer tube valve during fuel handling operations.

9.1.4.2.1.3 Spent Fuel Handling Machine

The spent fuel handling machine will be a refueling machine adapted for use in the spent fuel pool area. It will contain the same interlock features as described in Section 9.1.4.2.1.1, except as noted below for the Spent Fuel Handling Machine Translation Zone Interlock:

- A. Zone interlocks protect against running the load into walls or the gate of the storage area.
- B. If these interlocks fail, the spent fuel handling machine mast will protect the fuel assembly from damage in the event of wall or gate contact.

9.1.4.2.1.4 New Fuel Elevator

The following identifies and describes the functions of the interlocks that are part of the new fuel elevator. The new fuel elevator controls are located on the spent fuel handling machine control console.

A. New Fuel Elevator Hoist Cable-Slack Interlock

Stops the elevator motor should the cable become slack.

If this interlock fails, the operator can stop the elevator motion from the spent fuel handling machine console.

B. New Fuel Elevator Hoist Lock-Out Interlock

Prevents raising of the elevator with a fuel assembly in the elevator box. This interlock is a backup for the administrative control, which prohibits the placement of a spent fuel assembly in the new fuel elevator.

C. New Fuel Elevator Hoist Limit Interlock

Interrupts hoisting when "up" or "down" limits are reached.

9.1.4.2.1.5 CEA Elevator

The following identifies and describes the functions of the interlocks that are part of the CEA elevator.

A. CEA Elevator Hoist Cable-Slack Interlock

Stops the elevator motor should the cable become slack.

B. CEA Elevator Hoist Lock-Out Interlock

Prevents raising of the CEA elevator with a fuel assembly in the elevator support rack above the minimum safe water level for shielding water coverage. This interlock is a backup for the administrative control, which prohibits the placement of a spent fuel assembly in the CEA elevator.

9.1.4.2.1.6 CEA Change Platform

The following identifies and describes the function of the interlocks that are part of the CEA change platform.

A. CEA Change Platform Hoist Up-Stop Interlock

This interlock interrupts hoisting of a CEA assembly when the correct (full-up) vertical elevation is reached. A mechanical up-stop has also been provided to physically restrain the hoisting of a CEA assembly above the elevation which would result in less than the minimum shielding water coverage.

B. CEA Change Platform Hoist Overload Interlock

This interlock interrupts hoisting of a CEA if the load increases above the overload setpoint. The hoisting load is visually displayed so that the operator can manually terminate the withdrawal operation if an overload occurs and the hoist continues to operate.

C. CEA Change Platform Hoist Underload Interlock

Interrupts insertion of a CEA assembly if the load decreases below the underload setpoint. The insertion load is visually displayed so that the operator can manually terminate the insertion operation if an underload occurs and the hoist continues to operate.

9.1.4.2.1.7

Nuclear Annex
~~Fuel Building~~ Overhead Cranes

A. Cask Handling Hoist

The cask handling hoist is used to unload and transport new fuel shipping containers from the receiving bay area to the new fuel shipping container laydown area. It is also used for movement of the empty spent fuel cask from the truck/rail car unloading area to the cask laydown area and for the return of the loaded cask. The unloading area has

sufficient room to permit the cask to be upended with the cask transporter locked in place. The hoist is equipped with a continuously variable speed hoist controller.

The hoist accesses the spent fuel pool area during building construction to facilitate construction and the installation of the spent fuel storage racks. After the fuel racks have been installed, and before the fuel assemblies are placed in the racks, mechanical stops are installed on the bridge rails to prohibit the hoist from traveling over the spent fuel pool area.

The hoist is also provided with electrical interlocks to control bridge, trolley, and hoist travel and to minimize possible damage to the spent fuel shipping cask and the spent fuel pool during cask handling. The interlocks also prevent movement of the spent fuel cask over the new fuel racks. ^{coupled with the lifting equipment design and interlocks,} THE CASK HANDLING PROCEDURES RESTRICT THE HEIGHT THAT THE CASK CAN BE LIFTED TO ENSURE THAT THE BOTTOM OF THE CASK IS NEVER ABOVE THE OPERATING FLOOR. ^

B. New Fuel Handling Hoist

The new fuel handling hoist is used to move new fuel from the new fuel unloading area to the new fuel storage racks, the new fuel inspection stand, and the new fuel elevator. The hoist is provided with electrical interlocks to control the transfer path of the new fuel assemblies and to restrict fuel handling loads by either limiting the hoist motor stall torque or incorporation of load limiting devices. The hoist is restricted mechanically from allowing movement of new fuel over the spent fuel racks.

9.1.4.2.1.8

~~Reactor Building~~
Containment Polar Crane

^{TWO INDIVIDUALS} The polar crane is mounted on a circular crane wall and travels 360 degrees. The reactor building polar crane has a main hoist and an auxiliary hoist to handle the various loads during an outage. Provisions are made to ensure safe load handling. These provisions include automatic upper and lower hoist limits, overload limits, slow speed hoist operation, and a load handling path to prevent damage to any safety-related equipment from a heavy load drop. The polar crane is used to move the reactor vessel head and reactor vessel internals between the reactor vessel and various storage areas during outages, as described in Section 9.1.4.2.3.3. The polar crane hoist is able to operate at fast speed with an empty hook.

THIS RESTRICTION ON THE LIFT HEIGHT PRECLUDES PASSAGE OF THE SHIPPING CASK OVER THE NEW FUEL RACKS.

Figure 9.1-14 shows the lift rig in the configuration provided for removal of the upper guide structure assembly. In this configuration, the spreader beam supports three columns providing attachment points to the upper guide structure assembly. Attachment to the upper guide structure assembly is accomplished manually from the working platform. Correct positioning is assured by attached bushings that mate to the reactor vessel guide pins.

The clevis assembly, tie rod assembly, and spreader beam assembly which are common to this and the core support barrel lifting rig, are installed prior to lifting of the structure by the crane hook. The working platform also incorporates the holding fixtures for the extension shafts and CEAs.

9.1.4.2.2.8 Spent Fuel Handling Machine

The spent fuel handling machine as shown on Figure 9.1-23 is a refueling machine modified for use in the nuclear annex. The major differences are the longer bridge span and the interlocks to protect the equipment during movement within the pool, canal and cask laydown area.

9.1.4.2.2.9 New Fuel Elevator

A fuel elevator as shown on Figure 9.1-8 is utilized to lower new fuel from the operating floor to the bottom of the pool where it is grappled by the spent fuel handling tool. The elevator is powered by a cable winch and fuel is contained in a simple support structure whose wheels are captured in two rails. New fuel is loaded into the elevator by means of the new fuel ~~overhead crane~~ and new fuel handling tool.

HANDLING HOIST

A manually operated handwheel is provided for elevator operation in the event of a power loss.

9.1.4.2.2.10 Underwater Television

A closed circuit television system as shown on Figure 9.1-15 monitors the fuel handling operations within the refueling cavity. The camera is mounted on the refueling machine fuel hoist box (see Figure 9.1-4) so that the fuel assembly can be sighted prior to and during grappling and removal from the core. The system may also be used to initially align the refueling machine position indication system with the actual core location of the fuel assemblies. A monitor is provided at the refueling machine control console. The camera, if required for remote surveillance, or inspection, can be removed from its mount on the fuel hoist and handled separately. It is also used for core mapping after core loading before upper guide structure installation to confirm and record loaded core.

9.1.4.2.2.14 In-core Instrumentation and CEA Cutters

A portable underwater hydraulic cutter similar to that shown on Figure 9.1-16 is provided to cut the expended CEAs into lengths necessary to permit transfer to the spent fuel building in the transport container. A second cutter is used for disposal of the incore instrumentation leads.

9.1.4.2.2.15 Gripper Operating Tool

This tool is approximately seventeen feet long and consists of a two concentric tubes with a funnel at the end to facilitate engagement with the CEA extension shafts. When installed, pins attached to the outer tube are engaged with the extension shaft. The inner tube of the tool is then lifted and rotated relative to the outer tube which compresses a spring allowing the gripper to release, thus separating the extension shaft from the control element assembly.

9.1.4.2.2.16 Cask Handling Hoist

The cask handling hoist is ^{Before} capable of servicing the truck and rail car loading/unloading bay area, new fuel shipping container laydown area, the cask washdown area, the cask laydown area and the spent fuel pool. ~~After~~ ^{After} fuel assemblies have been placed in the spent fuel racks, mechanical stops are installed on the hoist bridge rails to prevent passage of the hoist over the spent fuel pool. The hoist has a minimum capacity of 150 tons and incorporates a variable speed hoist and electrical interlocks to control bridge and trolley travel. The major load handling paths of the cask handling hoist are shown on Figure 9.1-20.

9.1.4.2.2.17 New Fuel Handling Hoist

The new fuel handling hoist is capable of servicing the new fuel shipping container laydown area, the new fuel storage area, and the new fuel elevator. The hoist has a minimum capacity of 10 tons and incorporates electrical interlocks to control the transfer path of the new fuel assemblies and to restrict fuel handling loads. The hoist is mechanically restricted from passing over the spent fuel racks. The major load handling paths for the hoist are shown on Figure 9.1-20.

9.1.4.2.2.18 ^{Reactor Building} Containment Polar Crane

The reactor building polar crane is capable of servicing all major components requiring movement during a refueling outage. It incorporates, as a minimum, a 225 ton main hoist for handling the reactor vessel head and a 25 ton auxiliary hoist for lighter load handling. The hoist incorporates automatic upper and lower travel limits, overload limits and low speed operation to insure safe load handling. Crane control is from a trolley-mounted cab.

The major load handling paths for the crane are shown on Figure 9.1-19.

9.1.4.2.3 System Operation

9.1.4.2.3.1 New Fuel Transfer

After arrival of the new fuel shipping containers, the container covers are removed and the fuel assembly strongback raised to the vertical position and locked. The new fuel handling tool, attached to the new fuel handling hoist, is then locked to the fuel assembly, the fuel assembly clamping fixtures removed and fuel assembly removed from the shipping container. Next, the protective wrapping is removed and the fuel assembly is moved over to the new fuel storage racks where it is placed into its designated cavity. New fuel may be inspected by a new fuel inspection device before placement into the new fuel racks. The tool is unlocked from the assembly and the operation repeated until all assemblies have been placed in the racks. During initial core loading, all or a portion thereof of the new fuel assemblies may be placed into Region I of the spent fuel storage racks.

Prior to reactor refueling operations, the new fuel is removed from the new fuel storage racks and transferred to the new fuel elevator by using the ^{new} fuel handling ~~crane~~ ^{hoist} and the short fuel handling tool.

The new fuel elevator lowers the fuel assembly into the spent fuel pool to allow the spent fuel handling machine to transfer the fuel assembly to the spent fuel racks in Region I of the spent fuel pool or to the transfer system upender.

During reactor refueling operations, the new fuel assembly is placed in the upending mechanism, a spent fuel assembly is removed from the other position of the fuel carrier and transferred to a designated position in the spent fuel storage racks using the spent fuel handling machine and the spent fuel handling tool. The new fuel is then transferred to the reactor building.

9.1.4.2.3.2 Spent Fuel Transfer

Spent fuel transfer during refueling is discussed in Section 9.1.4.2.3.3.

The spent fuel handling machine transfers the spent fuel assemblies from the storage racks to the spent fuel shipping cask. This operation will be implemented when the spent fuel shipping cask loading pit is filled with spent fuel pool water and the gate between the spent fuel pool and the spent fuel shipping cask laydown area is opened. When the spent fuel

9.1.4.3

Safety Evaluation

9.1.4.3.1

NUCLEAR POWER
Fuel Building Overhead Cranes and Containment
Polar Crane

Reactor Building

The reactor building polar crane, the cask handling hoist, and the new fuel handling hoist are designed to prevent the drop of a heavy load such as the reactor vessel head or the spent fuel shipping cask. In addition, predetermined load paths for major lifts (see Figures 9.1-19 and 9.1-20), operator training, and regular crane maintenance minimize the possibility of load mishandling. The Owner-Operator's operating procedures will control the load paths and height of the reactor vessel closure head, the core support barrel and the upper guide structure above the pool floor.

COL APPLICANTS

Limit switches, electrical interlocks and mechanical interlocks prevent improper crane and hoist operations which might result in a fuel handling accident. This is also discussed in Section 9.1.4.2.1.7. The cask handling hoist is restricted from movement over the new and spent fuel storage areas when the fuel racks contain fuel assemblies. The new fuel handling hoist is restricted from movement over the spent fuel storage area when the spent fuel racks contain fuel assemblies.

The spent fuel cask laydown area is separated from the spent fuel pool by a gate and a structurally reinforced concrete wall. The gate is closed, sealed, and locked during all cask handling operations. The floor in the laydown area has been designed to withstand the impact of a shipping cask dropped from a height of 30 feet without breaching the integrity of the floor plate. The spent fuel pool gates are designed to open into the pool area so that they will be maintained closed by the water pressure when either the transfer system canal or the cask laydown area is drained.

Any small water loss as a result of local damage to the laydown area wall liner cannot be communicated to the spent fuel pool due to the closed gate and the integrity of the independent spent fuel pool liner. Damage to the gate is prevented during cask handling by stops on the bridge crane rail that limit cask travel and by the recessed gate design.

In accordance with the regulatory position of Regulatory Guide 1.13 and General Design Criterion 61 of Appendix A to 10 CFR 50, the hoists are also restricted from passing over the spent fuel pool cooling system for ESF systems which could be damaged by dropping the load.

Set points for the hoist interlocks are set to prevent falling or tipping of the loads into the fuel storage areas.

COL APPLICATION

Administrative controls prepared by the Owner-Operator preclude movement of heavy loads within the reactor building refueling cavity when the refueling machine contains a fuel assembly. During heavy load movement, the fuel transfer tube valve is closed to avoid water level changes in the spent fuel pool in the nuclear annex during postulated accident conditions such as dropping the heavy load on the reactor vessel pool seal.

Conform with the guidelines of

The nuclear annex hoists related to fuel handling and the reactor building polar crane ~~meet the intent of~~ Regulatory Guide 1.29, Positions C.1 and C.2 and Regulatory Guide 1.13, Positions C.1 and C.6 as they relate to the ability of the cranes to withstand the effects of earthquakes. With respect to radioactive release as a result of fuel damage, the cranes ~~meet the intent of~~ Regulatory Guide 1.13, Positions C.1 and C.2, and 57.1/ANSI-N208, ANS 57.2/ANSI-N210, NUREG-0554, and NUREG-0612.

9.1.4.3.2 Fuel Handling

Conform with the guidelines of

A failure modes and effects analysis is described in Table 9.1-2.

Direct voice and/or electronic communication between the control room and the refueling machine console and the spent fuel handling machine console is available whenever changes in core geometry are taking place. The communications will be independent of other communications channels to the control room. This provision allows the control room operator to inform the refueling machine operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Operability of the fuel handling equipment including the bridge and trolley, the lifting mechanisms, the upending machines, the transfer carriage, and the associated instrumentation and controls is assured through the implementation of preoperational tests and routines. Prior to the first actual fuel loading, the equipment is cycled through its operations using a dummy fuel assembly. In addition to the interlocks described in Section 9.1.4.2.1, the equipment has the following special features:

- A. The major systems of the fuel handling system are electrically interlocked with each other to assist the operator in properly conducting the fuel handling operation. Failure of any of these interlocks in the event of operator error will not result in damage to more than one fuel assembly.
- B. Miscellaneous special design features which facilitate handling operations include:

~~Comply with~~

to radioactive release as a result of fuel damage, the machines ~~meet the intent of~~ Regulatory Guide 1.13, Positions C.3 and C.5, ANS 57.1/ANSI-N208, ANS 57.2/ANSI-N210, NUREG-0554, and NUREG-0612.

9.1.4.3.3 Reactor Vessel Closure Head Handling

The reactor vessel closure head lift rig is designed, tested, and inspected to meet NUREG-0612 and the design criteria of ANSI N14.6. Analyses for the postulated head drops is performed to assure that the reactor vessel support system and shutdown cooling supply flow paths remain functional, that the core will remain in a coolable configuration, and that the k_{eff} of the core will remain below 0.95. The reactor vessel closure head drop is the worst case load drop accident.

The reactor vessel closure head lift rig and the reactor vessel internals lift rigs meets Regulatory Guide 1.29, Positions C.1 and C.2 and Regulatory Guide 1.13, Positions C.1 and C.6 as they relate to the ability of the lift rigs to withstand the effects of earthquakes. With respect to radioactive release as a result of fuel damage, the lift rigs ~~meet the intent of~~ Regulatory Guide 1.13, Positions C.3 and C.5, ANS 57.1/ANSI-N208, ANS 57.2/ANSI-N210, NUREG-0554, and NUREG-0612.

~~Comply with~~ conform with The
provisions of

9.1.4.4 Testing and Inspection Requirements

During manufacture of the fuel and CEA Handling Equipment at the vendor's plant, various in-process inspections and checks are required including certification of materials and heat treating, and liquid-penetrant or magnetic-particle inspection of critical welds. Following completion of manufacture, compliance with design and specification requirements is determined by assembling and testing the equipment in the vendor's shop. Utilizing a dummy fuel assembly having the same weight, center of gravity, exterior size and end geometry as an actual assembly, all equipment is run through several complete operational cycles. In addition, the equipment is checked for its ability to perform under the maximum limits of load, fuel mislocation and misalignment, and static and dynamic load conditions. All traversing mechanisms are tested for speed and positioning accuracy. All hoisting equipment is tested for vertical functions and controls, rotation, and load misalignment.

Hoisting equipment is also tested to 125% of specified hoist capacity. Fuel handling tools are proof tested to 150% of the maximum handling load. Setpoints are determined and adjusted and the adjustment limits are verified. Equipment interlock function, and backup systems operations are checked. Those functions having manual operation capability are exercised manually.

COL APPLICANT

Plant operating procedure guidelines will require appropriate operation training and crane inspections. The guidelines will also require the Operator-Owner to prepare operating procedures for preoperational load testing and checkouts of interlocks, brakes, hoisting cables, control circuitry and lubrication of fuel handling equipment.

ATTACHMENT 4

FAX

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9424 Files (w/o)
9612 files

DATE: October 12, 1993

NUMBER: OPS-93-0821

SUBJECT: Transmittal of Response to Action Items from the September 2, 1993 PRA Meeting

I am providing ABB-CE's response to two of the level 2 action items from the September 2, 1993 meeting between ABB-CE and the NRC on the PRA Open Items. If you have any questions on this information, please call me at (203)285-3926.

ACTION ITEM:

Provide a reference to a Westinghouse reassessment of temperature induced SGTR performed for the Swedish Ringhals plant.

RESPONSE:

The reference is: WCAP-11910; "Creep Rupture Failure of Primary Coolant Piping Prior to Reactor Vessel Failure for Severe Accident"; Lutz, R. J.; WPSD; July, 1988.

ACTION ITEM:

Provide an assessment of the contribution of temperature-induced SGTR events to core damage frequency and risk.

RESPONSE:

Temperature-induced SGTRs have been postulated to occur as a result of the natural circulation of very high temperature steam just prior to vessel during a high pressure severe accident sequence. Thus, temperature-induced SGTR events do not contribute to core damage frequency.

Releases due to temperature-induced SGTR events were not directly modeled in the System 80+ PRA because they were felt to be of low frequency and, provided a sustained failure of the Main Steam Safety Valves (MSSVs) does not occur, also of low consequence. A temperature-induced SGTR event occurs as a result of creep rupture of an SG tube due to the natural circulation of very high temperature steam from the core. SG tube failure occurs as a result of high temperature material creep at high pressure very near the time of vessel failure. A temperature-induced SGTR event will only occur if the RCS pressure is still at or near the Primary Safety Valve (PSV) setpoint at the time of vessel failure. This means that the RCS pressure at the onset of core damage is at the PSV setpoint and that the RCS is not depressurized prior to vessel failure.

For System 80+, Plant Damage States (PDSs) for which the RCS pressure was at the PSV setpoint at the onset of core damage have a total core damage frequency of $5.06\text{E-}07$ per year. In the System 80+ PRA, the probability for failure to depressurize the RCS prior to vessel failure using the RDS was calculated to be 0.2. Thus, the probability that the RCS pressure is at the PSV setpoint just prior to vessel failure is $1.01\text{E-}07$. In NUREG-1150, the probability of a temperature induced tube rupture given that the RCS pressure is at the PSV setpoint just prior to vessel failure was given as 0.014. More recent research performed by Westinghouse for the Ringhals plant (see reference in the previous response) indicates that the probability is closer to $1.0\text{E-}04$ to $1.0\text{E-}05$. Using the NUREG-1150 probability, the frequency for a temperature-induced SGTR is $1.42\text{E-}09$ per year. Using the

Westinghouse conditional probability value, the frequency for a temperature-induced SGTR is $1.01\text{E-}11$ per year.

As previously stated, if a temperature-induced SGTR occurs, it will occur just prior to vessel failure. At this time, almost all of the RCS inventory has been discharged into the IRWST through the PSVs. This includes most of the noble gases and the volatile Iodines and Cesiums. At the time of the SGTR, the RCS will begin to depressurize into the affected SG. The SG pressure will increase and the Main Steam Safety Valves (MSSV) will lift. This will result in a short release without a lot of noble gases or volatile Iodines and Cesiums. However, the vessel will fail shortly thereafter. This will result in a very rapid depressurization of the RCS and the affected SG. As the pressure in the affected SG decreases, the MSSVs will reseal. There would be a long term uncontrolled release only if one of the MSSVs failed to reseal. It is estimated that only the first bank of two MSSVs would lift to relieve the secondary pressure. Per data presented in the EPRI ALWR PRA Key Assumptions and Groundrules document, the probability that an MSSV will fail to reclose is $7.0\text{E-}03$ per valve. Thus, the total probability that an MSSV on the affected SG will fail to reseal is $2 * 7.0\text{E-}03 = 1.4\text{E-}02$. Therefore, the probability of an uncontrolled release due to a temperature-induced SGTR is $2.0\text{E-}11$ if NUREG-1150 data is used or $1.41\text{E-}13$ if the more recent Westinghouse data is used. This is about two orders of magnitude below the $1.0\text{E-}09$ filter value used in the System 80+ PRA.

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DATE: September 28, 1993

NUMBER: OPS-93-0798

SUBJECT: Transmittal of Response to Additional Question on ABB-CE's Responses to Question 6 Not Directly Related to DSER Open Items

I am providing ABB-CE's response to "Additional Question on ABB-CE's Responses to Question 6" not directly related to DSER Open Items, as documented in your fax of September 23, 1993. If you have any questions on this information, please call me at (203) 285-3262 or D. Finnicum at (203) 285-3926.

Additional Question on ABB-CE's Response to Question 6

1. In ABB-CE's Response to Question 6 regarding CSET end state frequency quantification it is stated that "this model (the containment spray flow mode) is version of PGOB01BX in which ... failure of CSAS was deleted because only manual actuation was credited." The fault tree model for containment spray flow is also attached in ABB-CE's response. However, an examination of the attached fault tree model shows that the requirement of CSAS has not been deleted (i.e., PFCN01A1 on Page 11 and PFCN01BX on Page 21 of PGOB01SA.CAF). Please clarify and explain this discrepancy.

2. The data presented in Table 19.12.1-4 of the updated PRA shows that the probability of losing containment heat removal (CHR) with containment spray (SC) available is extremely small. The ratio of CSS end state C (which is with CHR unavailable but with CS available) to CSS end state A (which is with both CHR and CS available) is about 0.0003. However, the results of the system (fault tree) analysis presented in Section 19.6.3.13 of the System 80+ PRA show that one of the most dominant contributor to system unavailability for containment heat removal (using containment spray system) is the common cause failure (CCF) of the containment spray heat exchanger outlet component cooling water (CCW) motor operated valves (MOVs). It contributes 29.2% to the total system unavailability. This seems to indicate that there is a fairly high probability of having CS but with CHR unavailable (due to loss of heat exchanger cooling by CCW) and does not seem to be consistent with the result presented in Table 19.12.1-4. Please explain this discrepancy. Also, please note that there is a Level 1 question on the failure probability used in the PRA for MOVs. Resolution of that issue may affect the results obtained here (another dominant contributor to the total system unavailability, 29.2%, is the CCF of the containment spray header isolation MOVs).

Response to Part 1 of Additional Question on ABB-CE's Response to Question 6

The CAFTA code is used to develop and quantify the fault tree models including PGOB01SA.CAF. One feature of this code is the ability to use logic flags to enable or disable (prune) certain "branches" of the fault tree. By setting the logic flag to "TRUE" the branch of concern will be included in the quantification, while by setting the logic flag to "FALSE" the branch will be pruned from the model and will not be included in the quantification.

On page 12 of fault tree model PGOB01SA.CAF, logic flags (AAALOCA, AASLOCA, & AAMLOCA) are included to enable or prune PFCN01A1 and PFCN01B1 from the model as desired. PFCN01A1 and PFCN01B1 portions of the fault tree model include failure of the actuation signal from CSAS trains A and B, respectively. In quantifying PGOB01SA.CAF, each of the logic flags was set to "FALSE" which causes gates PGOB18BX and PGOB20BX to be pruned from the model. Under this condition, only gate PGOB19BX feeds into gate PGOB16BX. Similarly, only gate PGOB21BX feeds into gate PGOB17BX. Thus CSAS was eliminated and only manual actuation of the Containment Spray System was credited.

Response to Part 2 of Additional Question on ABB-CE's Response to Question 6

The data presented in Table 19.12.1-4 of the System 80+ PRA was provided in response to NRC request and the values were not used in the quantification of the Plant Damage States

(PDSs). The quantification of the PDSs depends significantly on the status of the support systems and the transient or accident that lead to core damage. Given that core damage has occurred, any one of the six end states of the containment safeguard event tree can occur. Therefore, the probability of the six end states must sum to unity. End states "B" through "F" (Refer to Figure 19.12.1-1) were quantified using fault tree linking approach without considering the transient or accident that lead to core damage. End state "A" was quantified as the complement of the sum of end states "B" through "F". Since the containment safeguard mitigating systems share support systems with the core damage mitigating systems, the results presented in Table 19.12.1-4 are meaningless if the support systems are not properly accounted for. This is the case when the end states for the containment safeguard event tree are calculated without considering the transient or accident that lead to core damage. Therefore, the values presented in Table 19.12.1-4 are not used in quantifying the PDSs. In quantifying the Plant Accident Sequences (PASs) which are then mapped into the PDSs, fault tree linking approach was used to properly account for the common support systems utilized by the core damage mitigating systems and the containment safeguard systems.

End state "C" of the containment safeguard event tree represents failure of containment heat removal while containment spray flow is available. The probability of this end state is 3.0E-04 as presented in Table 19.12.1-4. The logic for the combinations of failures that lead to this conditions is shown as Figure 1 to this response. As shown in Figure 1, the ways in which containment heat removal can be lost but containment spray is available include common cause failure of the CCW outlet valves to the CSS heat exchangers or failure to remove heat from the individual trains. Loss of heat removal from the individual CSS train can result from failure of the CCW valve to the CSS heat exchanger or loss of flow in the CSS train. Based on Figure 1, the probability expression for loss of containment heat removal is as follows:

$$P(\text{CHR} * \text{CS}_{\text{ok}}) = P(\text{HXVCCF}) + P(\text{CS1}) * P(\text{HX2CCWV}) + P(\text{CS2}) * P(\text{HX1CCWV}) + P(\text{HX1CCWV}) * P(\text{HX2CCWV})$$

where,

$P(\text{CHR} * \text{CS}_{\text{ok}})$	=	Probability of loss of containment heat removal and containment spray is available
$P(\text{HXVCCF})$	=	Common cause failure probability of CCW valves to CSS heat exchangers (2.0E-04)
$P(\text{CS1}) * P(\text{HX2CCWV})$	=	Probability of loss of flow from CSS train 1 and Failure probability of CCW valve to CSS train 2 heat exchanger (3.2E-05)
$P(\text{CS2}) * P(\text{HX1CCWV})$	=	Probability of loss of flow from CSS train 2 flow Failure probability of CCW valve to CSS train 1 heat exchanger (3.2E-05)

$$P(HX1CCWV)*P(HX2CCWV) = \text{Failure probability of CCW valve to CSS train 1 heat exchanger and Failure probability of CCW valve to CSS train 2 heat exchanger (1.6E-05)}$$

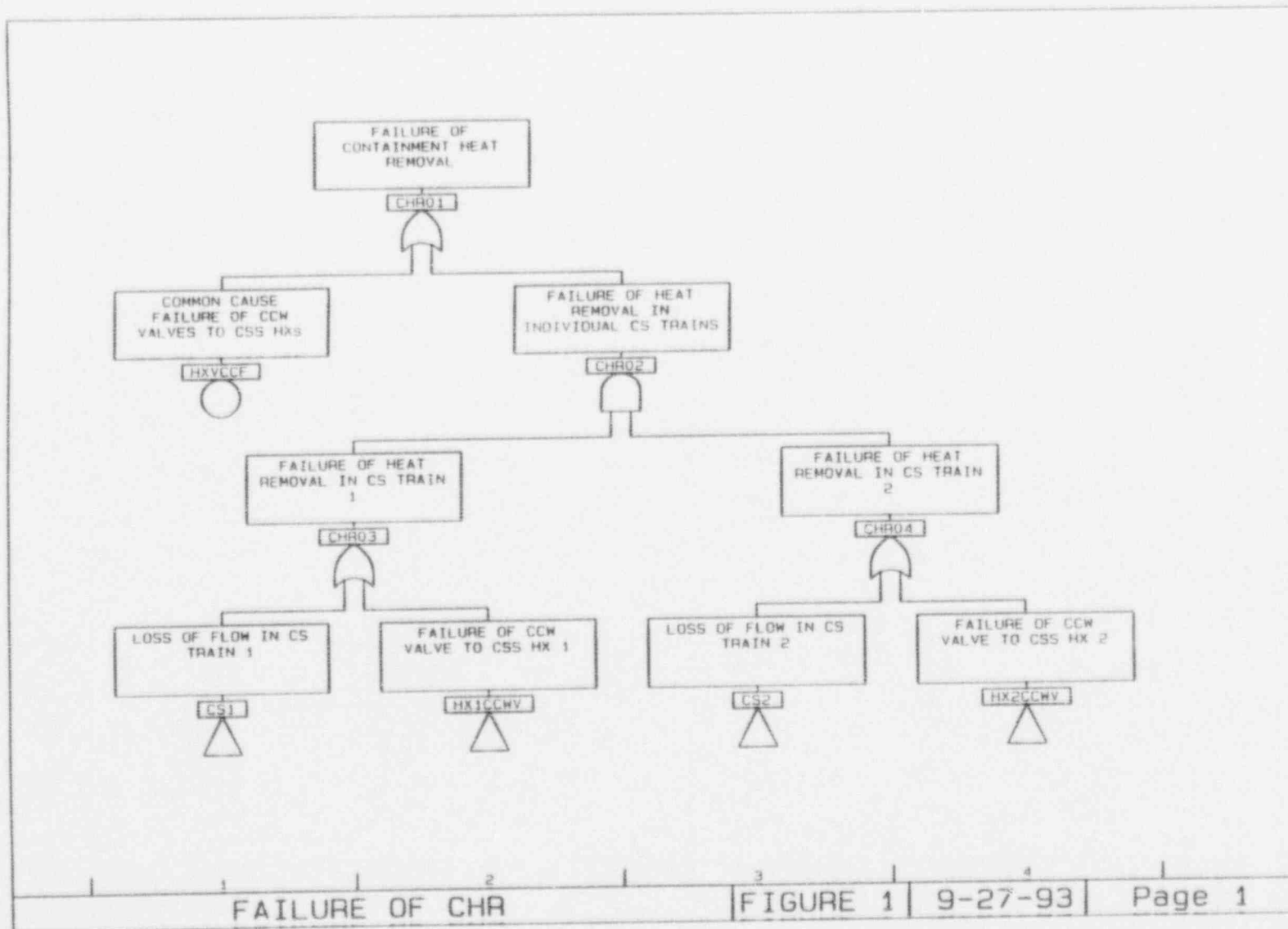
Figure 1 shows that loss of containment spray flow in both trains is a possible cutset. However, this cutset was eliminated from the above expression because it does not directly cause a loss of containment heat removal while containment spray is available. The values for the above probabilities were obtained from the fault tree analysis presented in Section 19.6.3.13. Substituting these values in the above expression, the probability for loss of containment heat removal and containment spray is available becomes:

$$\begin{aligned} P(CHR*CS_{ok}) &= 2.0E-04 + 3.2E-05 + 3.2E-05 + 1.6E-05 \\ &= 2.8E-04. \end{aligned}$$

This value is consistent with the value presented in Table 19.12.1-4 for end state "C".

The unavailability of the Containment Spray System (CSS) is due to loss of containment heat removal or loss of containment spray flow. The results of the fault tree analysis presented in Section 19.6.3.13 indicate that the conditional failure probability for loss of containment heat removal given that the CSS has failed is approximately 41%. (This probability was obtained by summing the individual probabilities for the cutset that lead to loss of containment heat removal and then dividing by the total CSS failure probability. This includes common cause failure of the CCW containment spray heat exchanger outlet valves.) Conversely, the conditional failure probability for loss of containment spray given that CSS has failed is approximately 59%. It is therefore meaningless to compare these values with the values shown for the end states of the containment safeguard event tree.

The question regarding the failure probability of MOVs used in the Level 1 PRA pertains to the Rapid Depressurization Valves (RC-406, RC-407, RC-408, RC-409). These valves will be tested less frequently than other types of MOVs. Therefore, the failure probability for the RDVs was increased to reflect an 18 months test interval (Refer to Table 19.5-2, sheet 22 of 22). The failure probability for other types of MOVs, including the CCW containment spray outlet valves, will remain unchanged because these MOVs will be tested on a quarterly basis.



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DATE: October 13, 1993

NUMBER: OPS-93-0824

SUBJECT: Transmittal of Response to Action Items from the September 2, 1993 PRA Meeting.

I am providing ABB-CE's response to an additional three of the level 2 action items from the September 2, 1993 meeting between ABB-CE and the NRC on the PRA Open Items. If you have any questions on this information, please call me at (203)285-3926.

ACTION ITEM:

Perform an additional sensitivity analysis to explore the effect of SDS Operation and hot leg/surge line failure on CCFP.

RESPONSE:

Sensitivity studies on the Effect of SDS operation and hot leg/surge line failure on CCFP were performed as part of the initial level 2 sensitivity analyses for the System 80+ PRA. They are documented in sections 19.14.1.6 and 19.14.1.7 of CESSAR-DC.

ACTION ITEM:

Clarify and confirm the appropriateness of ABB-CE's treatment of credit for scrubbing of revaporization releases by containment sprays in SGTR events.

RESPONSE:

With the containment sprays operating, containment pressure is maintained low prior to vessel breach. The RCS pressure will be higher than containment pressure because of the presence of the corium in the reactor vessel. At vessel breach, the hole in the vessel lower head will allow a steam flow from the higher pressure RCS to the lower pressure containment. The depressurization and high exiting steam flow from the RCS will lead to revolitization of fission products and carryout of these products into the containment. The fission products which revolitize after the vessel breach will tend to be swept into containment by the fact that the RCS will be at higher pressure than the containment due to the availability of containment heat removal. Once in containment, these revolitized fission products would be scrubbed from the containment atmosphere by the containment spray. Thus, the containment spray system is effective in scrubbing revolitization fission products for SGTR events. S80SOR, a modified version of ZISOR, was used to calculate all releases. S80SOR did not credit containment spray scrubbing of revolitization releases or any other releases for SGTR events.

ACTION ITEM:

Confirm the appropriateness of ABB-CE's assumptions that: (1) feedwater is available in virtually all SGTR PDSs retained in the level 2 analysis, and (2) the secondary side water level will always be maintained above the elevation of the SGTR.

RESPONSE:

- (1) The availability of feedwater was determined directly from the accident sequence definitions for all sequences that comprise a given PDS. Thus the
- (2) In responding to an SGTR event, ABB-CE's Emergency Procedure Guidelines call for isolating the ruptured SG when it is identified and conditions established such that the MSSVs will not lift. This will generally occur at about thirty minutes into the event. Once the ruptured SG has been isolated, the primary to secondary leakage will be the only source of inventory for the ruptured SG. Because the SG is isolated, the SG level will gradually increase. When the SG level reaches the high level setpoint, the procedures direct the operators to reduce the level to the low level setpoint by opening an ADV to steam the ruptured SG or by using the blowdown system to drain off some inventory. Both the high level and low level setpoints are above the top of the tube bundle, so for the case where the SG is isolated, the SG level will be above the elevation of the SGTR.

If an MSSV lifts and fails to reseal, or an ADV fails to reclose, the inventory in the ruptured SG will continue to be steamed to the atmosphere. If it is assumed that the operators had initiated isolation of the ruptured SG, there will be no EFW flow available for secondary inventory makeup to the ruptured SG. Thus, the primary to secondary leakage will be the only source of inventory makeup for the ruptured SG. With a stuck open valve, the ruptured SG pressure will be at or near atmospheric pressure. The primary to secondary leak rate will be high due to the high differential pressure between the primary system and the secondary system. For this scenario, even with the high primary to secondary leak rate, there is the possibility that the upper portion of the tube bundle would be uncovered. Thus, the probability that the rupture would be uncovered is a function of the location of the rupture. If the rupture is low in the tube bundle, it will be covered. If the rupture is at the top of the tube bundle, it would probably be uncovered. Thus, for this scenario, the secondary level will not always be above the elevation of the SGTR. To address this, the probabilities for the basic event SG-SCRUB in section 19.12.2.2.8.3.2.1 (page 19.12-97) have been redefined. For SGTRs with an isolation failure, the probability for SG-SCRUB will be set to 0.5. For SGTR events with isolation, the probability for SG-SCRUB is set to 1.0. SG-SCRUB is not applicable to other events. The marked up copy of page 19.12-47 is attached. As a result of this modification, the frequency for release classes RC4.4E, RC4.12E and RC4.18L will be cut in half and three additional release classes, RC4.22E, RC4.30E and RC4.36L, will be added. These new release classes will have the same frequency as the revised release classes

RC4.4E, RC4.12E and RC4.18L. This change will not affect overall risk because in calculating releases for SGTR events, S80SOR does not credit fission product scrubbing by any residual inventory in the ruptured SG. Thus, the releases for the new release classes will be the same as for the original release classes.

There are six plant damage states involving SGTR events that do not involve an unisolable leak in the ruptured SG (PDSs 181, 184, 193, 196, 218, and 220). In the CET, it is assumed that there is a potential for a containment isolation failure for all PDSs (Top event ISOL). For SGTR events, it was assumed that the isolation failure would be a stuck open MSSV. In the CET, "In-Vessel Fission product scrubbing" is modeled for both isolation failures and non-isolation failures using the same model. For the fraction of these PDSs without isolation failure, the value for SG-SCRUB should be 1.0, and for the fraction of these PDSs involving isolation failure, the value for SG-SCRUBB should be 0.5. However, the isolation status is not given in the definition of the PDS for the six PDSs (it is a function of the CET), and only one value can be set for the variable SG-SCRUBB for each PDS. To resolve this, six new PDSs will be created, PDS181A, PDS184A, PDS193A, PDS196A, PDS218A, and PDS220A. These PDSs will be identical to the original PDSs except that they will be defined to have an isolation failure. The CET basic event values will be changed such that the isolation failure probability will be set to 1.0, the PDS frequency will be reduced by a factor of $4.0E-02$, the original isolation failure probability, and the value for SG-SCRUBB will be set to 0.5. The CET basic event values for the original PDSs will be modified to set the isolation failure probability to 0.0, to reduce the PDS frequency by the frequency of the appropriate companion event, and to set the value for SG-SCRUBB to 1.0.

19.12.2.2.8.3.2**FP-SEC: Secondary Systems Scrub Fission Product Release**

This element defines the probability for secondary systems to scrub fission product releases from an otherwise intact RCS. Two potential sources of secondary scrubbing are considered. These are the residual secondary side water inventory above the SGTR break elevation (SG-SCRUB) or the water pool in the condenser (CONDEN-SCRUB).

19.12.2.2.8.3.2.1**SG-SCRUB: SG Water Level Above SGTR Elevation**

This element is defined as follows:

$P(\text{SG-SCRUB}) = 0.5$ for all SGTRs without the simultaneous loss of feedwater (this corresponds to virtually all SGTRs included in the PDSS)

$P(\text{SG-SCRUB}) = 1.0$ for all SGTRs with isolation

$P(\text{SG-SCRUB}) = 0.0$ for all other PDSSs

19.12.2.2.8.3.2.2**CONDEN-SCRUB: F.P. Release Via Condenser**

This event covers cases where the fission products are released from the SGTR to the atmosphere via the main condenser. If the fission products were released via this path, they could be scrubbed by water remaining in the condenser. This element is included for completeness only. Its probability is defined to be 0.0 for all PDSSs. Thus:

$P(\text{CONDEN-SCRUB}) = 0.0$ for all PDSSs.

19.12.2.2.9**Top Event 6: VRP: Vaporization Release Prevented**

This top event determines if the corium debris bed is coolable in the reactor cavity. If the debris bed is not coolable, it will react with the concrete in the cavity resulting in a vaporization release with high tellurium content. Core-concrete interaction is presumed to occur if the corium discharged from the reactor vessel remains in the cavity, and the core debris bed in the cavity is not cooled. This event was quantified using the supporting logic model presented in Figure 19.12.2-10. The basic events in this model are WETCAVITY and NOCOOLG. These elements are discussed in Sections 19.12.2.2.6.4.1.1.1 and 19.12.2.2.7.1.2.2.1.2.2, respectively.

19.12.2.2.10**Top Event 7: Revaporization Release Prevented**

This event determines if a revaporization release occurs late in the accident sequence significantly increasing the amount of

ATTACHMENT 5

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DATE: October 1, 1993

NUMBER: OPS-93-0784

SUBJECT: Supplemental Response to Second Follow-on Question For DSER Open Item 19.1.2.1.1.3-1

Per our discussions at our meeting on September 2, I am providing you with a supplemental response to the second follow-on question for DSER open item 19.1.2.1.1.3-1. This response provides the results of an evaluation of peak clad temperature for an 0.03 ft² medium LOCA. This analysis demonstrates that the peak clad temperature at the hot node in the core does not exceed 2200 degrees F. If you have any questions on this information, please call me at (203)285-3926.

DSER Open Item 19.1.2.1.1.3-1

Issue: Low-end of a Medium LOCA Break Size Sufficient for Decay Heat Removal Via the Break Only.

CE Fax #: OPS-93-0581 (Subject: Response to RAI on DSE's Open Item 19.1.2.1.1.3-1; Dated 7-22-93){*See also OPS-93-0631, Dated August 23, 1993*}

NRC Comments and Followup Questions: CE's transient analysis indicates that the reactor core is uncovered to a depth of approximately 6 inches for approximately 7 minutes. CE interprets this result as showing that adequate core heat removal can be achieved via the break so that secondary heat removal is not required. However, the staff believes that additional information is needed to make this determination. In PRA analyses, it is assumed that a core damage has occurred if part of the active fuel has been uncovered and a fuel cladding temperature of 2200 deg. F or higher is reached in any node of the core (see EPRI URD App. A, Vol. II, PRA Key Assumptions and Groundrules). Since a part of the active core is uncovered in this transient analysis, the next step should be to estimate clad temperatures vs time in order to determine whether or not core damage must be assumed in the PRA. The final determination should be made by taking into account, in addition to the maximum estimated clad temperature, also uncertainties in the calculations and models (part of best estimate computer code) as well as the values of the operating parameters used in the best estimate thermal-hydraulic transient analysis.

Supplemental Response: ABB-CE had performed a best-estimate transient analysis for a 0.03 ft² medium LOCA to demonstrate that secondary side heat removal via delivery of main or emergency feedwater was not required to prevent core damage. For this best estimate transient analysis, it was assumed that the normal control systems were in automatic and functioning properly immediately prior to the initiation of the transient. Therefore, the initial RCS pressure was 2250 psia, the initial pressurizer level was 26 ft., the initial steam generator level was 40.46 ft., and the initial steam generator pressure was 1000 psia. At T=0, an 0.03 ft² RCS break was initiated. Reactor trip occurred almost immediately. Main Feedwater was terminated on reactor trip and all emergency feedwater pumps were disabled. All but one of the safety injection pumps were also disabled. Thus, only one safety injection pump was available to respond to this accident. The analysis showed that the core uncovered slightly, starting at about 600 seconds. The uncovering lasted for about 1800 seconds with a maximum depth of uncovering of about 1.6 feet. Based on a review of the small LOCA analyses in section 6.3.3.3 of CESSAR-DC, ABB-CE concluded that there would be adequate cooling to prevent the peak clad temperature from exceeding 2200 degrees F at the hottest node in the core.

On September 2, 1993, ABB-CE and the NRC held a meeting in Windsor to discuss the System 80+ PRA. During this meeting, the NRC indicated that they felt there were enough differences between the small break LOCA cases presented in section 6.3.3.3 of CESSAR-DC and the 0.03 ft² medium LOCA case presented above that the NRC could not positively conclude that peak cladding temperature would not exceed 2200 degrees F during the core

uncovery. ABB-CE agreed to run an analysis of the peak cladding temperature for the 0.03 ft² medium LOCA.

In preparing to run the hot rod cladding temperature analysis, it was determined that a more conservative break location and scenario could be selected for the analysis. In the initial analysis, the selected break was a 0.03 ft² break in the bottom of the hot leg. For this break, two of the four RCPs were tripped and two were left running. The analyses presented in section 6 of CESSAR-DC show that cold leg breaks for which the RCPs are tripped at the time of reactor trip are limiting. Hot leg breaks are more advantageous because they have no injection flow spillage and better steam venting through the break. Keeping the pumps running provides a short term inventory distribution advantage by keeping the core supplied with inventory. Therefore, a new 0.03 ft² medium LOCA analysis was performed. This analysis was for an 0.03 ft² break in the bottom of the cold leg. Consistent with the chapter 6 analyses, all four RCPs were tripped at the time of reactor trip. It was assumed that the normal control systems were in automatic and functioning properly immediately prior to the initiation of the transient. Therefore, the initial RCS pressure was 2250 psia, the initial pressurizer level was 26 ft., the initial steam generator level was 40.46 ft., and the initial steam generator pressure was 1000 psia. Main Feedwater was terminated on reactor trip and all emergency feedwater pumps were disabled. All but one of the safety injection pumps were also disabled. Thus, only one safety injection pump was available to respond to this accident. This is conservative with respect to the section 6 analyses which credit two HPSI pumps for all medium break LOCAs except for a DVI line break for which only one HPSI pump is credited. For licensing LOCA analyses, the worst single failure, given the assumption of a loss of offsite power, is the failure of one diesel generator which disables two Injection pumps. Thus, for most LOCAs, two Injection pumps are available to respond to the LOCA. However, for a DVI line break, it is assumed that the break occurs in the DVI line for one of the two operating injection pumps and that the flow from that pump is lost out the break. Thus, only one Injection pump can deliver flow to the RCS. The DVI lines are 5 feet higher than the cold legs. Given that only one Injection pump is credited, the cold leg break used for this analysis is more limiting than the DVI line break because it is lower and thus results in more coolant inventory loss.

The results of this new 0.03 ft² medium LOCA analysis show that the core initially uncovers at about 1800 seconds and recovers at about 5200 seconds. The total uncovery time was about 3400 seconds. The maximum uncovery depth was about 2.5 feet. (See attached figure 3. In this plot, the 0 foot level is set as 1.5 feet below the bottom of the active fuel and the top of the core is at 14 feet.)

After the transient analysis was completed for the 0.03 ft² cold leg break, a hot rod cladding temperature analysis was performed to determine the peak fuel cladding temperature at the hot spot in the core. The following pages provide a description of that analysis and the results of the analysis. This analysis shows that the peak cladding temperature at the hottest node does not exceed 1100 degrees F.

ANALYSIS OF FUEL ROD CLADDING TEMPERATURE RESPONSE FOR AN 0.03 FT² SBLOCA

PURPOSE

This System 80+ analysis calculates the fuel rod cladding temperature response for a SBLOCA transient with a break area of 0.03 ft² in the RCP discharge leg. This SBLOCA is specified to be the smallest break size not dependent on auxiliary feedwater delivery to the steam generator secondary heat sink to mitigate the consequences of the accident with one ECCS safety injection delivery train available. The core pressure, reactor vessel two-phase mixture level, reactor vessel liquid mass, and the core power are used as boundary conditions for this hot rod peak cladding temperature (PCT) calculation. The purpose of this analysis is to show that the calculated fuel rod response for this transient does not exceed the ECCS Acceptance Criteria of 10CFR50.46.

COMPUTER CODES

The fuel cladding temperatures are calculated using the PARCH/REM computer code. PARCH/REM is part of the ABB-CE Small Break LOCA Realistic Evaluation Model or REM. The REM is designed for NSSS licensing calculations in the context of the September 16, 1988 revision of the ECCS Rule (10CFR50.46). The PARCH/REM code is a second-generation successor to the 1974 EM version of the PARCH code, specifically modified to perform realistic fuel rod temperature calculations for a SBLOCA transient. The REM version of PARCH is documented in CEN-373-P Vol. 1, which was submitted to NRC for review in April 1988, and is re-documented in CEN-420-P Vol.1 Pt.1, which was submitted to NRC for review in February 1993.

The PARCH/REM model evaluates on a closed channel basis the hot rod peak cladding temperatures and local oxidation percentages using boundary conditions supplied by a thermal-hydraulics NSSS model such as CEFLASH-4AS/REM or CENTS. The code solves the one-dimensional radial conduction equation for the fuel pellet, gap, and cladding at different axial positions along the fuel rod. The fuel rod model includes variable gap conductance as well as cladding swelling and rupture. The heat transfer correlations cover the forced convection and pool boiling regimes.

The mechanisms for convective heat transfer for the uncovered fuel rod are pool boiling below the two-phase fluid surface and convection and radiation to steam above the two-phase fluid surface. The mass flow rate of steam in the steam region is determined from the boiloff and flashing rates from the two-phase region computed for the average rod in the fuel rod assembly. Therefore, the steam flow rate calculated in PARCH/REM for cooling the hot rod represents the fuel assembly average steaming rate. An energy balance is performed over the steam region of the hot rod assuming no radial mixing and an axially uniform steam distribution. The cladding temperature during the boiloff time period is calculated using a three region radial conduction model for the fuel pellet, gap, and cladding.

The core pressure, reactor vessel two-phase mixture level, reactor vessel liquid mass, and the

core power boundary conditions are provided by the CENTS thermal-hydraulic NSSS systems transient simulation code. CENTS is an interactive simulator for PWRs based on ABB CE's best-estimate, realistic, and licensing codes. CENTS is used for a wide variety of transients and plant conditions ranging from operational maneuvers through accidents and severe core uncover. CENTS is documented in a series of topical reports and technical users manuals, as CENPD-282-P, and was submitted in 1992 for NRC review as a licensing tool.

INPUT PLANT PARAMETERS AND CONDITIONS

The input plant parameters and conditions for PARCH/REM are calculated using the Limit Value Approach (LVA). The LVA is documented in CEN-373-P Vol. 2, which was submitted to NRC for review in December 1988. In the LVA, plant conditions and input parameters with a significant effect on peak cladding temperature (PCT) are set simultaneously at their Limit Value. The Limit Value is that value in the range of a parameter which maximizes the PCT. The resulting Limit Value PCT is higher than a PCT calculated with nominal (realistic) plant parameters. The Limit Value PCT includes all the uncertainty due to plant conditions and input parameters by assuming all the adverse-bound values of these parameters occur simultaneously instead of assuming a statistical distribution (within their range) of the values of these parameters. The following plant conditions and input parameters are represented using LVA:

Core Power - 3992 MWt. The core power Limit Value is the rated thermal power plus 2% power to allow for calorimetric uncertainty.

Peak Linear Heat Generation Rate - 14.7 kw/ft. This value of PLHGR is increased by 1 kw/ft over the specified maximum value (13.7 kw/ft) to cover uncertainty in the representation of the hottest rod in the core.

Decay Heat - 1979 ANS + 2 Sigma. A two sigma increase applied to the nominal values of the decay heat fraction provides an upper bound of greater than 95% probability.

Pin-to-Box Ratio - 1.0811. This input parameter sets the power of the assembly containing the hot rod for defining the steam flow rate during core uncover. This is the value for the time-in-life occurrence of peak power for the highest powered fuel rod.

Axial Power Shape - ASI = -0.3. This value represents the most negative axial shape index allowed. The most adverse axial power shape has the maximum pin peak power at the highest elevation uncovered during the transient. In the SBLOCA analysis, this axial distribution is transformed, over time, into a somewhat flatter axial distribution which is representative of decay heat production during the core boiloff period of the transient. This Limit Value approach to defining the axial power shape produces an axial peaking factor of 1.629 at the 85% elevation from the bottom of the core.

The boundary conditions for PARCH/REM are calculated for a 0.03 ft² break in the bottom of the cold leg using the CENTS code. The input plant parameters and conditions for CENTS are nominal in all aspects except for the decay heat which, as described above, is conservatively taken as the 1979 ANS decay heat plus two sigma. To conservatively bound this scenario, the RCPs are tripped at the same time as reactor trip and steam generator isolation. For cold leg break locations, this leads to the most limiting depth and duration of core uncover compared to other break locations where RCPs are allowed to continue running. To further conservatively bound this scenario, only one HPSI pump delivery is allowed. For S80+, HPSI delivery is by direct vessel injection. Therefore, even with single failure conditions, two HPSIs would normally be credited for the cold leg break transient.

RESULTS

The results of this PARCH/REM calculation are as follows:

Peak Cladding Temperature = 1051 °F

Peak Local Cladding Oxidation Percentage = 0.299 %

Peak Core-Wide Cladding Oxidation Percentage = < 0.058 %

The cladding temperature transient for the hot spot (Node 21, 100% elevation from bottom of core) is shown in Figure 1. Figure 2 shows the cladding temperature transients for the nodes below the hot spot (Node 17, 80% elevation, Node 18, 85% elevation, Node 19, 90% elevation, and Node 20, 95% elevation). The core pressure, reactor vessel two-phase level, reactor vessel liquid mass, and the core power used as boundary conditions are shown in Figure 3.

As described above, the Limit Value PCT result includes the uncertainty due to plant parameters and plant conditions. This temperature of 1051°F (including the uncertainties) is well below the 2200°F limit on PCT prescribed by the Acceptance Criteria of 10CFR50.46. Similarly, the peak local cladding oxidation percentage of 0.299% is well below the 17% limit prescribed by 10CFR50.46. The REM uses the hot rod oxidation results as a conservative indication of the core-wide oxidation. Based on this assumption, the peak core-wide oxidation is less than the value based on the hot rod. The peak core-wide cladding oxidation percentage of <0.058% is well below the 1% limit prescribed by 10CFR50.46.

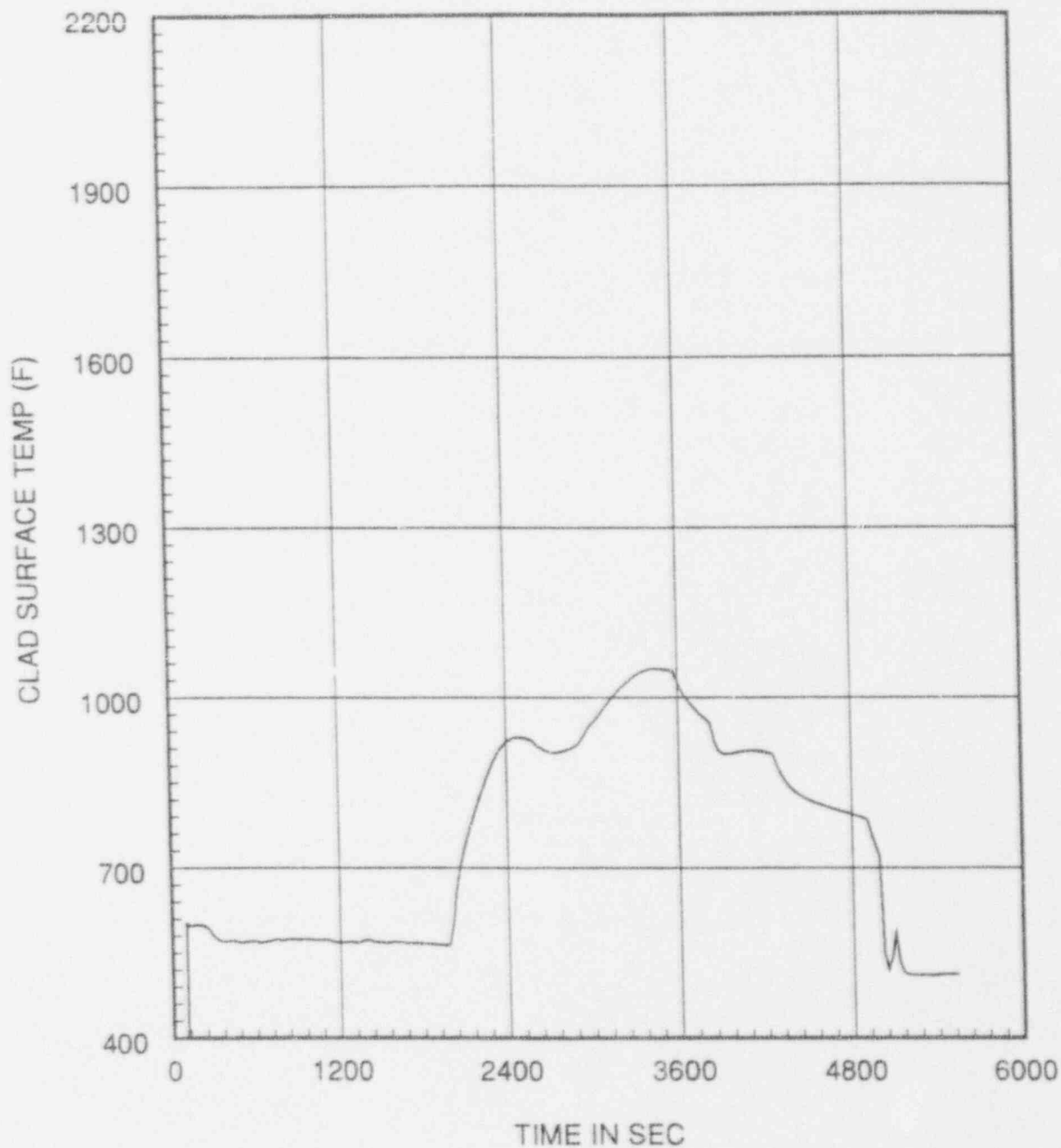
FIGURE 1
SYS80+ SBLOCA 1 SI PUMP COOLING
CLADDING SURFACE TEMP
NODE 21

PRM.1.1

(Hot Spot)

0u17jm89

1993/09/28



SYS80+ SBLOCA 1 SI PUMP COOLING
CLADDING SURFACE TEMP
NODE 17

FIGURE 2

SYS80+ SBLOCA 1 SI PUMP COOLING
CLADDING SURFACE TEMP
NODE 18

PRM 1.1

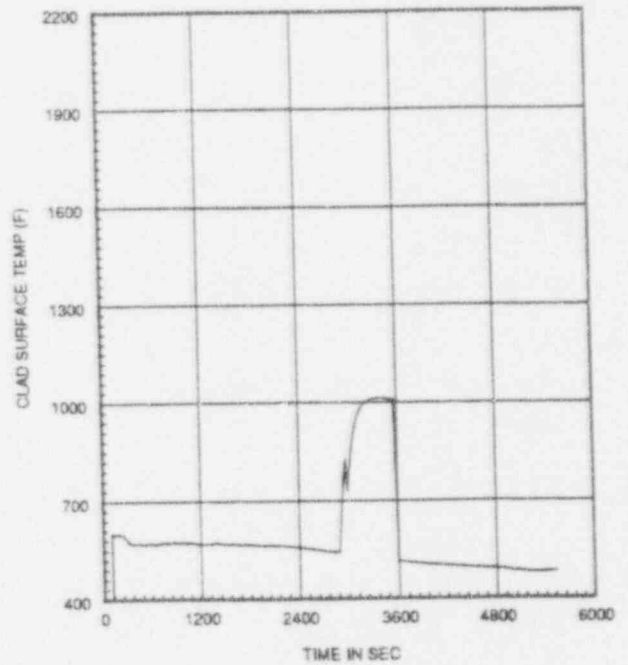
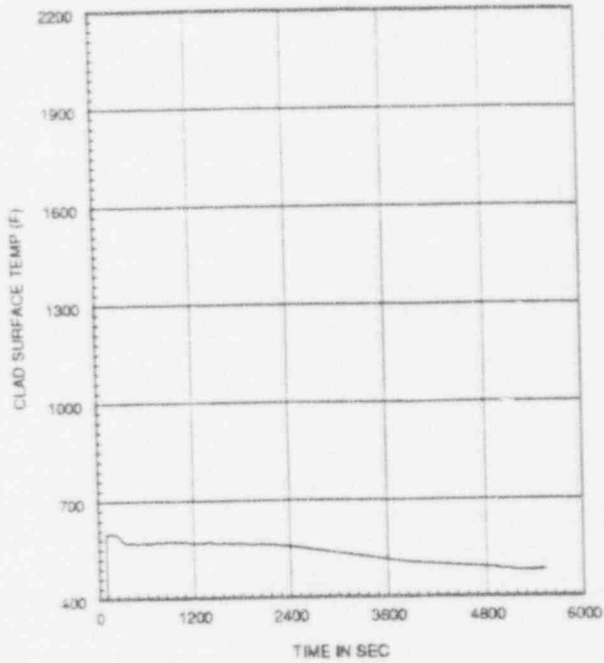
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PRM 1.1

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1993/08/28



SYS80+ SBLOCA 1 SI PUMP COOLING
CLADDING SURFACE TEMP
NODE 19

SYS80+ SBLOCA 1 SI PUMP COOLING
CLADDING SURFACE TEMP
NODE 20

PRM 1.1

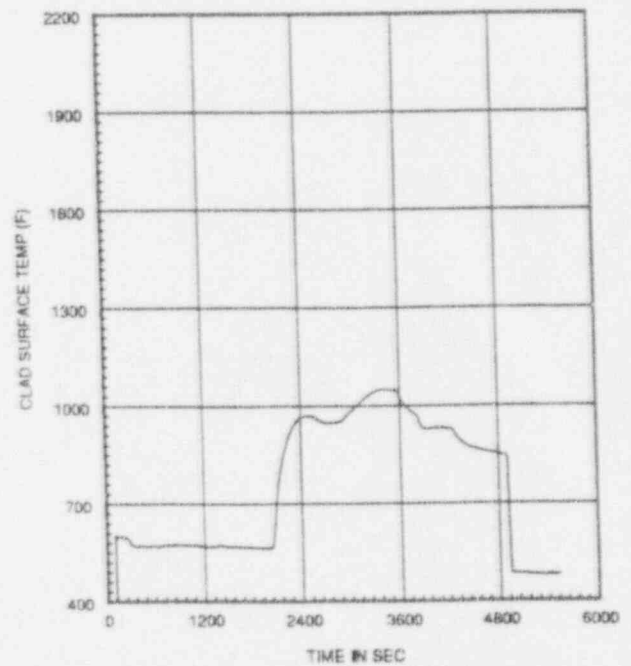
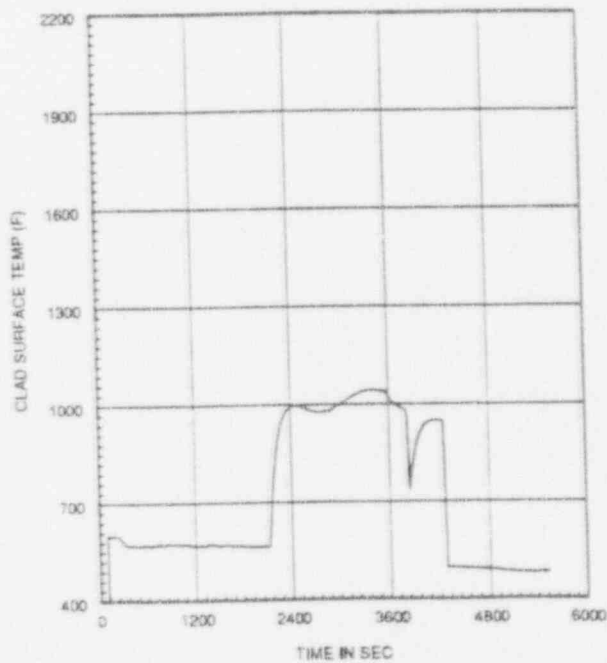
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PRM 1.1

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1993/08/28



SYS80+ SBLOCA 1 SI PUMP COOLING
CORE PRESSURE
BOUNDARY CONDITION

FIGURE 3

SYS80+ SBLOCA 1 SI PUMP COOLING
REACTOR VESSEL MIXTURE LEVEL
BOUNDARY CONDITION

CENTS80120

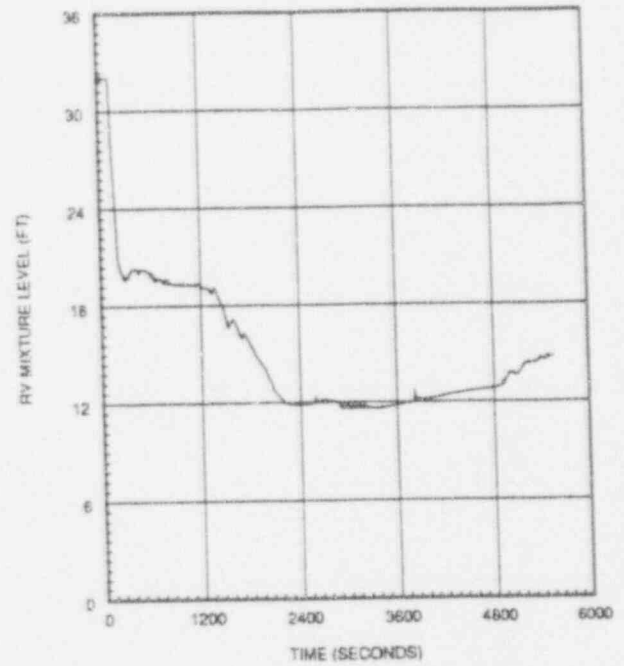
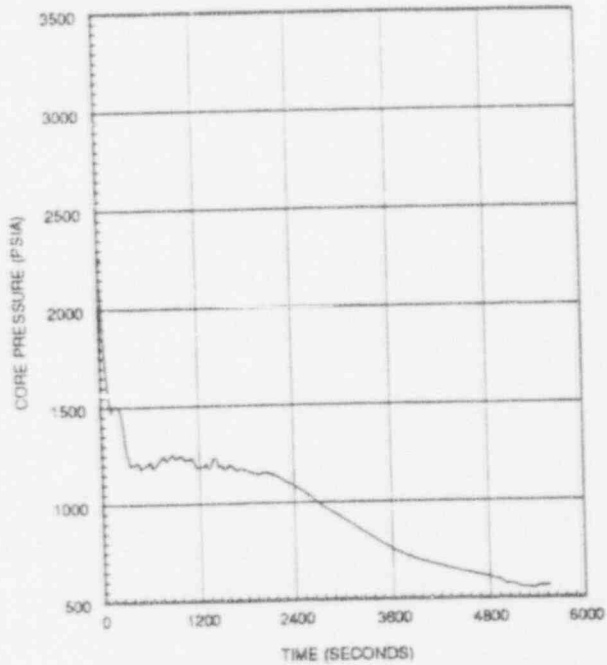
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SYS80+ SBLOCA 1 SI PUMP COOLING
REACTOR VESSEL LIQUID MASS
BOUNDARY CONDITION

CENTS80120

ABCDEFGHIH

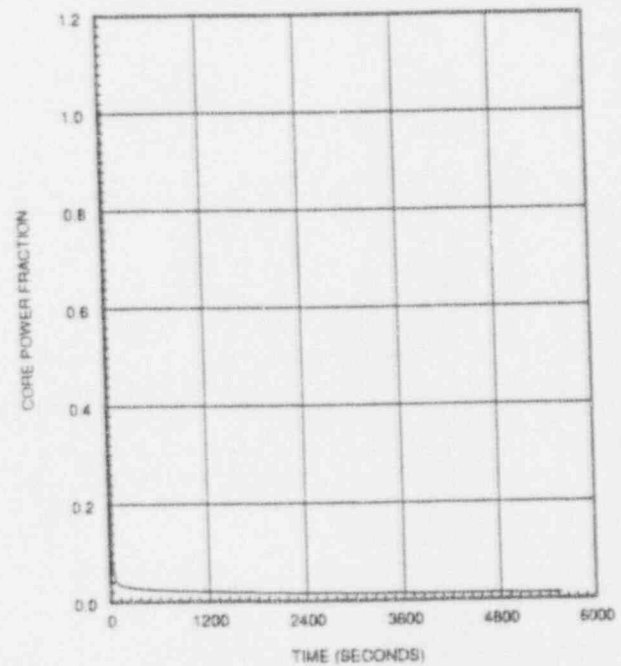
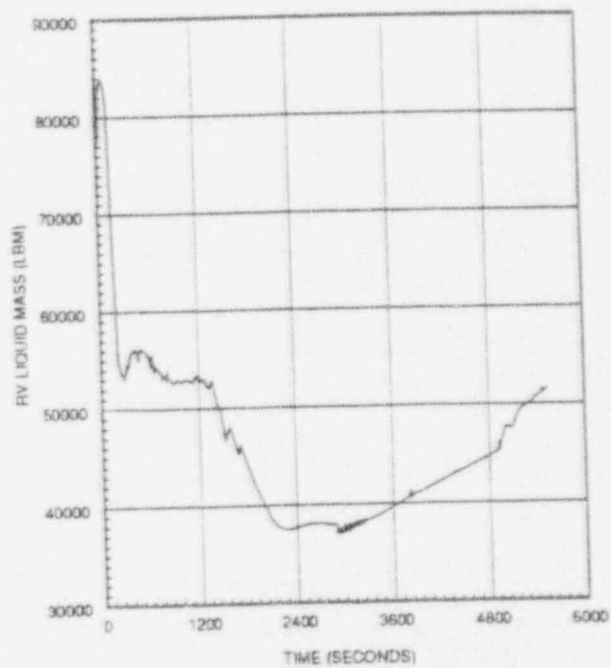
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SYS80+ SBLOCA 1 SI PUMP COOLING
NORMALIZED CORE POWER
BOUNDARY CONDITION

CENTS80120

ABCDEFGHIH

2706403



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9424 Files
9612 files

DATE: October 7, 1993

NUMBER: OPS-93-0814

SUBJECT: Transmittal of Supplemental Response to Follow-on Question for DSER Open Item 19.1.2.1.1.3-2

Per our discussions at our meeting of September 2, I am providing a supplemental response to the follow-on question to DSER Open Item 19.1.2.1.1.3-2 as documented in your fax of June 14, 1993. This supplemental response provides additional details on the initial conditions for the transient analysis and some additional timing information. If you have any questions on this information, please call me at (203)285-3926.

DSER Open Item 19.1.2.1.1.3-2

NRC Comments and Follow-up Questions:

This open item concerns the feasibility of using the damaged steam generator for ASC to reduce RCS pressure below the SCS pump shutoff head before the core is uncovered. It also requires an assessment of the radioactivity release through the ADV during the cooldown period.

The applicant cited a few references in the updated System 80+ PRA (References 9-11 of System 80+ PRA) to argue that if ASC is initiated within about 15 minutes of an SGTR event, the core will remain covered and the SCS can successfully provide RCS inventory control. The applicant also used the analysis results performed in the PRA for a small LOCA event as additional justification. As for the radioactivity release, the applicant used the results in CESSAR-DC for a SGTR event with a loss of offsite power and stuck open ADV to argue that the radiological release for the present case would be within 10CFR100 limits.

Although the accident events whose analysis results are used in the above arguments are not the same as that of interest here, they do seem to provide some relevant information for the issues raised here. However, as can be seen from the referenced small LOCA analysis (Figure 1 of Response to Open Item 19.1.2.1.1.3-1), the water level in the core region drops to near the top of the active fuel soon after the RCPs are tripped. The top of the active fuel remains barely covered throughout the rest of the transient even after the contents of the SITs are discharged (at RCS pressure of about 600 psia and $t=1100$ seconds). This may not be a big concern for a small LOCA event, but for an SGTR event, significantly larger radioactivity may be released directly to the environment if part of the core is uncovered, even for a short period of time. Furthermore, the performance of a damaged SG may not be as effective as an intact SG, the ASC used in an SGTR event may not be as effective as that in a small LOCA event, and it is likely that much longer time will be required for RCS pressure to drop to 600 psia when the SITs can start to inject. The main concern, therefore, is that whether the core can remain covered for the whole time period. To eliminate the above concerns, an analysis specifically prepared for SGTR is desired. We also noted that, based on figure 3 (of the Response to Open Item 19.1.2.1.1.3-1), core exit fluid temperature drops from 565 °F to 415 °F in about 17.5 minutes after ASC is started. This raises the question about whether the reactor will become critical again.

The success of ASC for SGTR requires the operator to properly diagnose the need of ASC and initiate ASC actions within about 15 minutes of accident initiation. The probability used in the updated System 80+ PRA for the operator to fail to perform ACS for SGTR is 0.07. Although SGTR recovery procedures were mentioned in the PRA, details of the procedures (or whether the procedures had already been prepared), are not provided. Since ASC involves actions that are contradictory to standard actions required for SGTR recovery (e.g., isolation), and also involves the opening of a release path of core radioactivity to the environment, the SGTR recovery procedure must provide sufficient information for the operator to make a correct and timely decision, consistent with the failure probability used in the PRA. This point should be emphasized in the preparation of the procedure. From the data presented in the updated PRA, ASC seems to play a very important role in the

determination of the total CDF of SGTR events. The leading SGTR sequence (SGTR-17) involves the failure of ASC. Its frequency of $2.73\text{E-}7/\text{year}$ constitutes 95% of total SGTR CDF ($2.87\text{E-}7/\text{year}$).

Response: ABB-CE provided an initial response to this follow-on question on August 26, 1993 via fax number OPS-93-0641. This response presented the results of a transient analysis which demonstrated that if an aggressive secondary cooldown was initiated within 15 minutes of an SGTR with failure of Safety Injection (SI), the RCS could be successfully depressurized to the point at which the SCS pumps could be used for injection without uncovering the core. On September 2, 1993, ABB-CE and the NRC met in Windsor to discuss the System 80+ DSER PRA open items. At that time, the NRC requested that ABB-CE provide additional detail on initial conditions for the transient analysis and some additional detail on the timing of the transient.

A best estimate transient analysis was performed to demonstrate that ASC could be successfully accomplished for an SGTR. For this analysis, it was assumed that the normal control systems were in automatic and functioning properly immediately prior to the initiation of the SGTR. Thus, normal operating parameters were at or near their normal setpoints. Initial pressurizer pressure was 2252 psia and initial pressurizer level was 26.2 feet. Initial steam generator pressure was 1017 psia and initial pressurizer level was 39.5 feet. The initial cold leg temperature was 567 degrees F and the initial hot leg temperature was 621 degrees F. At time zero, a double ended guillotine break of one Steam Generator (SG) tube was initiated. All four SI pumps were disabled and both charging pumps were disabled.

At 771 seconds, the reactor tripped on low pressurizer pressure (1825 psia.). Within 5 seconds, the main feedwater ramped back to 5% flow with flow being delivered to both steam generators. The Turbine Bypass Control System opened the Turbine Bypass Valves to deliver steam from both SGs to the condenser. At 780 seconds, the pressurizer drained. At 1200 seconds, a manual MSIS was generated and the steam lines from both generators were isolated. This terminated Turbine Bypass steam flow. At this time one ADV on each steam generator was opened. Five percent feedwater flow was delivered to both steam generators. (This is equivalent to the flow from one EFW pump per generator.) At 1240 seconds, two RCPs were manually tripped per procedures due to low RCS pressure ($< 1200\text{psia}$). At 1270 seconds, Reactor Vessel Upper Head (RVUH) void formation began and at 1670 seconds, the hot leg saturated. At 1680 seconds, the RVUH was empty and the reactor vessel node began to void. At about 2400 seconds the lowest level in the reactor vessel, approximately 28 feet, was reached. At this time the RCS pressure reached 600 psi and the SITs began to inject. All four SITs were assumed to be available for injection. The reactor vessel level was recovered and subcooling in the hotleg was regained at about 2600 seconds.

The radioactivity released during the transient was calculated using the standard chapter 15 dose calculation methodology and the conservative assumptions used for the SGTR dose calculation for chapter 15. The calculated 2 hour GIS thyroid dose was 15 Rem and the whole body dose was 0.585 Rem. The calculated 2 hour PIS thyroid dose was 43.7 Rem and the PIS whole body dose was 0.5 Rcm.. Both of these calculated releases are well within the 10CFR100 limit of 300 Rem..

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DATE: October 11, 1993

NUMBER: OPS-93-0818

SUBJECT: Responses to Requests for Additional Information (RAIs) Related to the PRA-Based Seismic Margins Analysis (SMA)

I am providing you with ABB-CE's response to RAI 8 related to the PRA-Based Seismic Margin Assesment as documented in your fax of August 31, 1993. If you have any questions on this information, please call me at (203)285-3926.

Question 8: It is stated in the submitted analysis that although the HCLPFs for some SSCs were assessed to be less than 0.6g, a design commitment is made by CE to achieve HCLPF values of at least 0.6g. It is clear that a failure to achieve this goal would result in a plant HCLPF lower than 0.6g. Please indicate how this goal can be achieved. For each SSC that a design commitment is made, please provide a brief description of the physical parameters and other characteristics that can be changed, as well as the way they can be changed, to achieve the goal. For example, in recent meetings with the NRC staff, CE indicated that the design of the steel containment vessel will be modified to provide positive connection between the vessel and the concrete base as well as the internal concrete structures. Please describe the failure mechanism that limits the steam generator supports HCLPF value to 0.6g. Are there any measures being considered to prevent this failure mechanism and thus increase the HCLPF value? Please use the results and insights drawn from this analysis as a potential input to ITAAC, as applicable.

Response: In the initial guidance on Seismic Margins Assessments (SMAs) for advanced light water reactors provided by the NRC staff, the staff indicated that a plant HCLPF of 0.6g was the required goal for advanced light water reactors. (Note: The commission recently issued a policy statement that said that ALWRs would be required to demonstrate a plant HCLPF of 1.67 times the design basis earthquake of 0.3g. Thus, the plant HCLPF required for ALWRs is 0.5g.) As indicated in the first SMA question of your fax of August 31, 1993, existing SMA guidance accepts the adequacy of the CDFM HCLPF when the CDFM approach is used. In performing the SMA for System 80+, ABB-CE used the CDFM approach for calculating HCLPF values for structures and components. The HCLPF values for the systems and the overall plant HCLPF were calculated using the min-max approach consistent with use of the CDFM HCLPF values. It is ABB-CE's intent to demonstrate compliance with the NRC's HCLPF goal for advanced light water reactors based on the CDFM HCLPF values and all design commitments were based on CDFM HCLPF values. Because ABB-CE wanted to maintain the capability to extend the PRA-based SMA to a full seismic PRA in the future, HCLPF₅₀ values were calculated from the CDFM HCLPF value for each component and structure and the results of the SMA were presented in terms of the HCLPF₅₀ values. The HCLPF₅₀ values are a factor of 1.2 lower than the CDFM HCLPF values.

The dominant seismic failure mode for the containment was seismically induced slipping/overturning. ABB-CE made a design commitment to achieve a HCLPF of at least 0.6g by providing a positive connection between the containment shell and the concrete embedment. It was intended that this commitment was to achieve a CDFM HCLPF of 0.6g. Three design alternatives, including Nelson studs and shear bars, were considered. Preliminary evaluations indicated that all three options would provide a CDFM HCLPF of greater than 0.6g for seismically induced slipping/overturning of the containment shell. The option that was selected was to install shear bars on the containment shell. The CDFM HCLPF calculated for the containment structure with the addition of the shear bars is 0.73g. The HCLPF₅₀ value is 0.61g. Thus, the design commitment was met by addition of the shear bars.

The initial HCLPF₅₀ value calculated for the steam generator supports was 0.56g . This was based on a CDFM HCLPF of 0.67g. The dominant failure mode was seismically induced failure of the SG snubber lever assembly. The calculated HCLPF value was based on the strength of a snubber assembly procured for an existing System 80 plant. The HCLPF value has been recalculated based on additional information for the SG snubber. The most recent CDFM HCLPF calculated for the SG is 0.73g. The HCLPF₅₀ value is 0.61g. Thus, the SG supports meet the design commitment.

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DATE: October 13, 1993

NUMBER: OPS-93-0825

SUBJECT: Transmittal of Response to Follow-on Question for DSER
Open Item 19.1.2.3-1

I am sending you ABB-CE's response to the follow-on question for DSER Confirmatory Item 19.1.2.3-1. This response is consistent with our discussions of 10/5/93. If you have any additional questions on this item, please call me at (203)285-2958.

- D. The concept of defense in depth applies to shutdown modes as well as Mode 1. The more ways that the operator can maintain coolant inventory and remove decay heat, the lower the risk. The presence of SIS capability in shutdown is an example of added defense in depth.

Loss of DHR Insights

- INSERT
6
- A. Reduced inventory is the most critical operation. The operator should be aware of this and plant activities should be scheduled accordingly. Use of nozzle dams is encouraged as a method of limiting the time spent in this mode.
- B. The operator must have procedures and training to align the SCS train to the IRWST and use it to makeup inventory or do a feed and bleed operation.
- C. Failure of the standby SCS train for either DHR or feed operation is dominated by failure of control valves and MOVs. An aggressive valve testing and maintenance program on the SCS and CSS would reduce shutdown risk.
- D. The use of the CVCS system to makeup inventory is an important recovery action in reduced inventory operation. It also acts as a temporary (about 12 hours) cooling technique. The operator should have procedures and training on its use.
- E. Safety injection for feed and bleed is an important DHR method in shutdown modes. Having two of the four SIS trains available during most shutdown modes is an important new technical specification.
- F. The CSS pumps are designed as an installed spare to the SCS. The operator must have procedures and training on its alignment and operation. Again, valve maintenance and testing is important for shutdown risk reduction.

LOCA Insights

- A. SCS feed is an important makeup method for a LOCA. Training and procedures are required.
- B. The dominant failure mode for SCS feed is failure of control valves and MOVs. A valve maintenance program is important.
- C. Safety injection using the SIS is important for LOCAs. Since manual actuation is required, training and procedures are required.
- D. The CVCS system is another important makeup method that should be available with training and procedures.

DSER Open Item 19.1.2.3-1

Issue: Submit Risk Analysis For Shutdown And Low Power Operation

CE Submittal: Amm. M to CESSAR; Includes Response to staff RAIs on the risk analysis for shutdown and low power operation and submittal of revised analysis; Dated 3-15-93

NRC Comments And Follow-up Question:

Question 5. The shutdown flood analysis addresses only the floods that are the result of LOCAs. With the reduced redundancy and the impact of shutdown related activities on the operability of flood barriers, floods from sources other than the primary system should be incorporated into the shutdown flood analysis.

REVISED RESPONSE 5. As noted in the assumptions (Page 19.8-39), great care has been taken in the design of the plant to minimize flood potential. Secondary flooding sources located in the turbine building are confined to that building. Station service water and component cooling water heat exchangers are located outside of the nuclear annex. These and other features limit the volume of water available in the nuclear annex. These features have been added to Section 19.15, assumptions and insights.

The following insights (Insert 6) will be added to the general insights in Section 19.8.1.2:

- F. There are two trains of SCS and it is important that the COL applicant maintains a configuration management system for maintenance activities on the SCS and its support systems. Configuration control is important because all plant risks, all accidents and incidents, and all accident precursors arise because of critical configurations which have occurred. If configurations were managed so that critical, high-risk configurations did not occur, then the risks would be small and accidents or incidents would occur rarely. Table 19.8.3.1-4 (developed from the more extended dependency Table 19.6.1-1) is an example of the systems that support each SCS train. The COL applicant should identify the systems, structures and components (SSCs) that support DHR (as well as other safety functions). The COL applicant should consider the overall effect of removing SSCs identified above from service on the DHR safety function. The COL applicant should limit normal maintenance on combinations of equipment so that an additional single or common cause failure would not cause total loss of DHR. A configuration management system should help to insure the availability of the standby SCS.

- G. If one train is lost because of fire, flood, or random component failure, it is important that the other train have the highest possible availability. The COL applicant should develop procedures and a configuration management strategy to handle the period of time when one of the two DHR paths is unavailable. In this case (a technical specification violation) the operator should suspend the maintenance and testing activities on equipment that support the operating SCS train. Given failure of one train, the operator should restore any systems that support the other train and are out for maintenance.

TABLE 19.8.3.1-4

EXAMPLE OF SYSTEM DEPENDENCIES FOR THE SHUTDOWN COOLING SYSTEM

SSC TRAIN 1

4.16 KV SAS201
 480 V MCC TRAIN A
 480 V MCC TRAIN C
 125 VDC SA0801
 CCWS TRAIN 1A or 1B
 SIS TRN 1 (DVI VALVES)
 IRWST (SOURCE, INJEC.)
 IRWST (COOLING CAPAB.)
 CSS TRAIN 1 (PUMP*)

(SUPPORTING CCWS 1A)

4.16 KV SAS201
 125 VDC SA0801
 480 V MCC TRAIN X
 SSWS TRAIN 1A or 1B

(SUPPORTING CCWS 1B)

4.16 KV SCS210
 125 VDC SC0801
 480 V MCC TRAIN X
 SSWS TRAIN 1A or 1B

(SUPPORTING SSWS 1A)

4.16 KV SAS201
 125 VDC SA0801

(SUPPORTING SSWS 1B)

4.16 KV SCS210
 125 VDC SC0801

DG1 or SBAC

SSC TRAIN 2

4.14 KV SBS201
 480 V MCC TRAIN B
 480 V MCC TRAIN D
 125 VDC SB0801
 CCWS TRAIN 2A or 2B
 SIS TRN 2 (DVI VALVES)
 IRWST (SOURCE, INJECTION)
 IRWST (COOLING CAPAB.)
 CSS TRAIN 2 (PUMP*)

(SUPPORTING CCWS 2A)

4.16 KV SBS201
 125 VDC SB0801
 480 V MCC TRAIN Y
 SSWS TRAIN 2A or 2B

(SUPPORTING CCWS 2B)

4.16 KV SDS201
 125VDC SD0801
 480 V MCC TRAIN Y
 SSWS TRAIN 2A or 2B

(SUPPORTING SSWS 2A)

4.16 KV SBS201
 125 VDC SB0801

(SUPPORTING SSWS 2B)

4.16 KV SDS201
 125 VDC SD0801

DG2 or SBAC

* The CSS pump can be used interchangeable with the SCS pump.

- K. Flood protection is also incorporated into the Component Cooling Water Heat Exchanger Building and Station Service Water Structure. These structures are divisional separated by walls such that a flood in one division cannot flood the other division.

These design features assure that the risk associated with floods is minimal.

In the shutdown mode, the initiating events and associated core damage scenarios which are the dominant contributors to risk are the Loss of Decay Heat Removal and Loss of Coolant Accident scenarios addressed in Sections 19.8.4.1 and 19.8.4.2, respectively.

Based on the design features described above, the separation and flooding protection afforded individual shutdown system trains is such that the dominant flooding risk scenario is one in which the flood originates in the operating train of the SCS. This event causes both a loss of the decay heat removal and a loss of RCS inventory.

The frequency of core damage associated with all LOCA scenarios, as calculated in Section 19.8.4.2, was found to be $1.35\text{E}-7$ per calendar year. Using the LOCA frequency information provided in BNL Coarse Screening Analysis²¹, it was determined that approximately 61% of the H, J, and K LOCA events originated in the SCS. Applying this information to the LOCA core damage frequency from Section 19.8.4.2, the flooding core damage frequency contribution is approximately $8.2\text{E}-08$.

→
INSERT 7

The following section (Insert 7) will be added to Section 19.8.4.4, Page 19.8-41.

NON-LOCA FLOOD SOURCES DURING SHUTDOWN OPERATION

The System 80+ design emphasizes the elimination and minimization of potential flood sources within safety-related areas as a means of flood protection. The station service water system and the component cooling heat exchangers used to remove decay heat from the RCS are located outside the Nuclear Annex. The condenser Circulating Water System is also located outside the Nuclear Annex. The location of these major sources of water to outside the Nuclear Annex reduces in-plant cooling water to a limited volume which can be easily accommodated to limit the extent of flooding during shutdown operations. The water systems within the Nuclear Annex are closed systems with a define volume of water. These water systems are located below elevation 70+0, with the exception of the Chilled Water System which is located at elevation 70+0. A major flood protection design feature of the System 80+ design is the divisional wall below elevation 70+0 which prevents flood waters from migrating to the opposite division. This prevent a flood in one division from affecting the other division of safety systems. There are no openings in the division wall.

The non-LOCA internal flood sources that were identified during power operation are also the sources within the Nuclear Annex during shutdown operations. The potential flood sources and the description of the flood zones are described below.

The following internal flood sources were determined to have the potential for release within the Nuclear Island:

Flood Source Volume

Component Cooling Water System (CCWS)	24,700 ft ³
Incontainment Refueling Water Storage Tank (IRWST)	72,958 ft ³
Emergency Feedwater System (EFWS)	46,785 ft ³
Fire Protection System (FPS)	80,203 ft ³
Chemical and Volume Control System (CVCS)	<u>161,075 ft³</u>
TOTAL VOLUME	385,721 ft ³

The volumes of the internal flood sources were determined base on the following:

- CCWS - This volume is the estimated volume of water contained in one division of CCWS
- IRWST - This volume is based on the normal operating water volume of 545,800 gallons from CESSAR-DC Table 6.8-1.

- EFWS - This volume is based on the volume of water contained in one EFW tank, 350,000 gallons, from CESSAR-DC Table 10.4.9-1.
- FPS - This volume is base on the volume of the fire protection water supply tanks, 600,000 gallons, as given in CESSAR-DC Section 9.5.1. It is assumed the contents of the FPS piping does not significantly add to the volume of water contained in the tanks due to preaction valves which limit the amount of water contained within the system piping.
- CVCS - This volume is based on the combined estimated volume of the Holdup Tank (525,000 gallons), Boric Acid Storage Tank (180,000 gallons), and the Reactor Makeup Water Tank (500,000 gallons). Actual internal volumes of these tanks as given in CESSAR-DC Table 9.3.4-4 are less than or equal to the estimated volumes. The volume of the water contained within CVCS piping was considered insignificant as compared to the combined volume of water in the tanks since it is highly unlikely that a pipe break in the CVCS would cause all three tanks to simultaneously drain to the Nuclear Island.

The following are the flood zone free volumes based on the general arrangement drawing of elevation 50+0 and assuming 50% of each flood zone's total volume is taken up equipment, internal walls, structures, etc.

Flood Zone
Volume

- I. Diesel Generator Area (DGA)
35,496 ft³
- II. Control Complex Area (CCA)
134,071 ft³
- III. Fuel Handling Area (FHA)
236,560 ft³
- IV. Reactor Building Subsphere Area (RBSA)
61,512 ft³
- V. Emergency Feedwater Pump Room (EFWPR)
5,014 ft³
- VI. Chemical and Volume Control System Area (CVCSA)
284,109 ft³

Areas identical to the above flood zones due to symmetry were not called out as separate flood zones since they have the same free volume and internal flood sources (i.e., DGA, CCA, RBSA).

To further illustrate the free volume of the Nuclear Island below elevation 70+0 by division, the volumes for each division are listed and totaled below:

Division I Free Volumes

DGA	35,496 ft ³
CCA	134,071 ft ³
FHA	236,560 ft ³
RBSA	61,512 ft ³
EFWPR (2)	<u>10,028 ft³</u>

Total 477,667 ft³

Division II Free

DGA	35,496
CCA	134,075
CVCSA	284,109
RBSA	61,512
EFWPR (2)	<u>10,028</u>

Total 525,216

A description of each flood zone and the potential internal flood sources for each zone are described below:

Flood Zone I - Diesel Generator Area (DGA)

This zone contains one of the two emergency diesel generators and its associated diesel generator support systems.

The Component Cooling Water System is considered the only potential internal flood source for this zone since CCW is supplied to the diesel generator support systems within the DGA and is present during all modes of operation. Fire Protection System water is excluded from the DGA by preaction valves located outside the DGA flood barrier boundary. DGA flood barrier integrity prevents other flood sources from entering the DGA.

Comparing the free volume of the DGA (35,496 ft³) and the volume of the CCWS flood source (24,700 ft³) shows that the volume of flood water is much less than the free volume of this flood zone. Therefore, should there be a CCWS pipe break within the DGA, the resulting flood water will be contained within the affected DGA below elevation 70+0 and the emergency diesel generator in the opposite division will be unaffected by this flood.

Flood Zone II - Control Complex Area (CCA)

This zone contains the vital electrical distribution equipment, vital batteries, vital instrumentation, and Instrument Air System equipment.

The Component Cooling Water System is considered the only potential flood source for the CCA since CCW is supplied to the instrument air compressors with the zone and is present during all modes of operation. Fire Protection System Water is excluded from the CCA by preaction valves located outside the CCA flood

barrier boundary. CCA flood barrier integrity prevents other flood sources from entering the CCA.

Comparing the free volume of the CCA (134,071 ft³) and the volume of the CCWS flood source (24,700 ft³) shows that the volume of flood water is significantly less than the free volume of this flood zone. Therefore, should there be a CCWS pipe break within the CCA, the resulting flood water will be contained within the affected CCA below elevation 70+0 and the electrical equipment in the opposite division will be unaffected by this flood due to the integrity of the divisional wall.

Flood Zone III - Fuel Handling Area (FHA)

This zone includes a portion of the Reactor Building Subsphere and the CVCS equipment area in Division I. Equipment located in this zone includes a CVCS charging pump and miniflow heat exchanger, a containment spray pump and heat exchanger, a safety injection pump, the containment cooler condensate pumps and tanks, the CVCS chemical addition package, two component cooling water pumps and various MCCs, MUXs, and panelboards.

The following are potential internal flood sources for the FHA zone:

- CCWS - The CCWS is a large source of water present in the FHA, during all modes of operation. Based on the flood barrier arrangement and flood zone figure, the CCW pumps and suction lines are located within the FHA. In the event of a break in this moderate energy piping system within the FHA, the contents of one division of the CCWS has the potential to empty and drain to elevation 50+0.
- EFWS - The EFW tank in this division is a large source of water present in the FHA during all modes of operation. In the event of a break in this moderate energy piping system within the FHA, the volume of the EFW Tank in this division has the potential to empty and drain to elevation 50+0.
- FPS - The FPS is a potential source of water which could enter the Nuclear Annex through FPS piping located in the FHA. In the event of a break in the FPS piping within the FHA the volume of the two fire protection water supply tanks has the potential to empty and drain to elevation 50+0.
- IRWST - Should there be a break in the Containment Spray System or the Safety Injection System piping located in the Subsphere quadrant located within this flood zone, the contents of the IRWST has the potential to empty and drain to elevation 50+0.

Comparing the FHA flood zone free volume (236,560 ft³) to each of the above potential internal flood source volumes shows that the FHA free volume is considerably larger than any of the flood source volumes. Therefore, flood water from either of the applicable flood sources will be contained within the flood zone below elevation 70+0 and the equipment in the opposite division will be unaffected by the flood.

Flood Zone IV - Reactor Building Subsphere Area (RBSA)

This zone excluded the turbine-driven emergency feedwater pump room. Equipment located within this zone include a shutdown cooling system pump, heat exchanger, and miniflow heat exchanger, a safety injection pump, electrical panels, MUXs, and MCCs.

The following are potential internal flood sources for the RBSA zone:

- IRWST - Should there be a break in the Shutdown Cooling System or the Safety Injection System piping located in this subsphere quadrant, the contents of the IRWST has the potential to empty and drain to elevation 50+0
- FPS - The FPS is a potential source of water which could enter the Reactor Building Subsphere through FPS piping located in the RBSA. In the event of a break in the FPS piping within the RBSA, the volume of the two fire protection water supply tanks has the potential to empty and drain to elevation 50+0.

Comparing the free volume of the RBSA (61,512 ft³) to the internal flood source volumes for the IRWST (72,958 ft³) and the FPS (80,203 ft³) reveals that the water volume of both flood sources exceeds the free volume of the RBSA. Therefore, each of the flood sources will completely fill the flood zone. Although the RBSA is shown to completely fill with the flood sources, flood water is restricted to the flood zone by the flood barriers leaving the remaining three subsphere quadrants unaffected.

Flood Zone V - Emergency Feedwater Pump Room (EFWPR)

This zone contains a motor-driven emergency feedwater pump. The motor-driven EFW pump room was chosen as a flood zone since it is somewhat smaller in size to the turbine-driven EFW pump room. This flood zone will be considered typical of the four EFW pump rooms.

The following are potential internal flood sources for the EFWPR:

- EFWS - The EFWS suction lines are located in each EFW pump room creating a path for the EFW tanks to empty into the room should there be a line break in this piping.

- FPS - The FPS is considered a potential source of water which could flood the EFW pump room due to the presence of FPS piping within this flood zone.
- CCWS - Component Cooling Water is provided to the motor-driven EFW pumps and is therefore present within this flood zone during all modes of operation.

Comparing the free volume of the EFWPR (5,014 ft³) to the internal flood source volumes shows that this flood zone will completely fill with each flood source considered. However, the flood barriers will confine flood waters to the room. The remaining EFW pump in the division as well as the two EFW pumps in the opposite division will be unaffected.

Flood Zone VI - Chemical and Volume Control System Area (CVCSA)

This flood zone contains much of the CVCS equipment including one of the CVCS charging pumps and its miniflow heat exchanger. Also included with this zone are the division II CCW pumps, a containment spray pump and heat exchanger, a safety injection pump, various electrical panels, MCCs, and MUXs.

The following are potential flood sources for the CVCSA:

- CCWS - The CCWS is a large source of water present in the CVCSA, during all modes of operation. Based on the flood barrier arrangement and flood zone figure, the CCW pumps and suction lines are located within this flood zone. In the event of a break in this moderate energy piping system within the zone, the contents of one division of CCWS has the potential to empty and drain to elevation 50+0.
- EFWS - The EFWS Tank in this division is a large source of water present in the CVCSA during all modes of operation. In the event of a break in this moderate energy piping system within this flood zone, the volume of the EFW Tank in this division has the potential to empty and drain to elevation 50+0.
- IRWST - Should there be a break in the Containment Spray System or the Safety Injection System piping located in the Subsphere quadrant within this flood zone, the contents of the IRWST has the potential to empty and drain to elevation 50+0.
- PFS - The FPS is a potential source of water which could enter the Nuclear Annex through FPS piping located in the CVCSA. In the event of a break in the FPS piping within this flood zone the volume of the two fire protection water supply tanks has the potential to empty and drain to elevation 50+0.

CVCS - The volume of the CVCS Tanks is assumed to be a potential source of water which could enter the Nuclear Annex through CVCS piping located throughout the CVCS Area, including upper elevations. For conservatism it is assumed the volume of the three largest tanks of the CVCS (Holdup Tank, Boric Acid Storage Tank, and Reactor Makeup Water Tank) are combined and drained to elevation 50+0 in the CVCS Area.

Comparing the free volume of the CVCS Area ($284,109 \text{ ft}^3$) to each of the potential internal flood source volumes shows that the flood source volumes are significantly less than the free volume of this flood zone. Therefore, flood water from either of the applicable flood sources will be contained within the flood zone below elevation 70+0 and the equipment in the opposite division will be unaffected.

In looking at the estimated total free volume of each System 80+ division below elevation 70+0 (approximately $477,667 \text{ ft}^3$ in division I and $525,216 \text{ ft}^3$ in division II), and comparing it to the total volume of the potential flood sources ($385,521 \text{ ft}^3$), it is evident that the Nuclear Island is of sufficient size to contain any of the postulated internal flood sources within a single division below elevation 70+0.

The frequency of loss of decay heat removal during shutdown operation is based on operating experience⁽²¹⁾ and includes all events that lead directly to loss of decay heat removal. A review of the events reported in Reference (21) shows that there were no non-LOCA flooding events that caused loss of decay heat removal. Based on zero events and approximately 1000 years of Pressurized Water Reactor (PWR) operation through the end of 1992, the estimated frequency of a non-LOCA flood is $7.0\text{E}-04$ per year. Assuming an average capacity factor of 70% for all PWRs, the estimated frequency of non-LOCA flood during shutdown operation is approximately $2.3\text{E}-03$ per year ($7.0\text{E}-04 \div 0.3$). This frequency is small in comparison to the total frequency (0.12 per year⁽²¹⁾) for loss of decay heat removal. Several design features are incorporated into the System 80+ design to prevent non-LOCA flooding that would cause a loss of decay heat removal during shutdown operation. These include the location of major safety related flooding sources outside the Nuclear Annex and the divisional wall that separates redundant safety related equipment. The major safety related flooding sources include the Station Service Water System and Component Cooling Water heat exchangers. All water systems within the Nuclear Annex are of limited volume that can be contained within the flood zone. Because there are two redundant divisions of SCS and the divisions are separated by the divisional flood wall, a non-LOCA flood would affect only the SCS division in which the flood originated.

The COL applicant will provide a detailed flood analysis to verify the assumptions and results of this preliminary analysis.

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DATE: October 14, 1993

NUMBER: OPS-93-0826

SUBJECT: Transmittal of Information on CENTS

At our meeting on October 4, 1993, Mr. Ruben asked that ABB-CE provide some information on the CENTS code. The CENTS (Combustion Engineering Nuclear Transient Simulator) code is the code used for verifying selected success criteria for the System 80+ PRA. CENTS was selected because it is ABB-CE's best-estimate realistic transient analysis code. CENTS is a real time-interactive computer code that can simulate NSSS behavior for a wide range of conditions, from normal operation to severe accidents with significant two-phase, including core uncover. Models are provided for the core, the primary system, the secondary system and the control systems. In general, in terms of complexity and completeness, the CENTS models are comparable to those in the industry's advanced plant simulation and design codes. CENTS is the basis for ABB-CE's full scope plant simulators. It has been benchmarked against the CEFLASH-4AS code and against plant data. I am attaching a brochure and several papers which describe CENTS and some of its applications. If you have any questions on this information, please call me at (203)285-3926.

CENTS

COMBUSTION ENGINEERING NUCLEAR
TRANSIENT SIMULATION CODE

C E N T S

Combustion Engineering Nuclear Iransient Simulator

ABSTRACT

The CENTS Nuclear Plant Analyzer (NPA) is an interactive simulator for nuclear power plants having a pressurized water reactor (PWR) design. CENTS is based on Combustion Engineering's best-estimate design and licensing codes. It combines the high fidelity and first-principles rigor of these codes with a user-friendly interactive environment that makes it easy to learn and use on computer engineering workstations. The code provides real time simulation with live graphics, sophisticated probing and control capabilities for the user, and extensive but flexible input/output and display interfaces.

CENTS is designed to be a versatile, high-fidelity simulation tool for use by the nuclear power industry's engineering, operations and training organizations. It may be used to simulate a wide variety of transients and plant conditions, ranging from operational maneuvers through multiple malfunctions, accidents and severe core uncover.

CENTS applications include realistic engineering analyses for emergency response drills, procedures development and training; assessment of revisions to design, technical specifications, controls and instrumentation; full-scope simulator upgrades and validation. CENTS is also ready for submittal for NRC approval, for licensed safety analyses.

TABLE OF CONTENTS

<u>Section</u>	<u>Title</u>	<u>Page</u>
	ABSTRACT	1
1.0	EXECUTIVE SUMMARY	1
2.0	CODE DESCRIPTION	2
	2.1 Background	2
	2.2 Models	2
	Reactor Core Neutronics	2
	Primary Reactor Coolant System	4
	Protective and Control Systems	5
	Steam Generator Secondary Coolant System	6
	2.3 Features and Benefits	7
3.0	ACQUISITION MODES AND SYSTEM REQUIREMENTS	9
4.0	DOCUMENTATION	10
TABLE 1	Key Features of the CENTS Nuclear Plant Analyzer	11
TABLE 2	Examples of the Transient Capabilities of CENTS	16
Figure 1	CENTS Model of Two-Loop Pressurized Water Reactor	17
Figure 2	Three-Dimensional Radial Neutronic Nodes	18
Figure 3	Three-Dimensional Radial Thermal Hydraulic Nodes	19
Figure 4	Primary System Geometric Model for a 4 Loop Westinghouse Plant	20
Figure 5	Steam Generator Secondary System Geometric Model for a Two Loop C-E Plant	21

1.0 EXECUTIVE SUMMARY

One of the major lessons learned in the past decade is that nuclear power plant personnel need a versatile engineering tool that can accurately represent the plant for a variety of applications related to design, training, operation, and event mitigation. The CENTS Nuclear Plant Analyzer (NPA) is an interactive, high-fidelity, user friendly, complete nuclear power plant simulation code for use by utility engineering, operations and training personnel.

CENTS runs interactively faster than real time on a desktop computer workstation. It features a full set of simulation software, and is based on models developed by C-E for its design codes. State-of-the-art mathematical methods enable the design models to run fast while maintaining high fidelity and to accommodate a wide spectrum of design basis events. CENTS provides a higher degree of fidelity than existing simulation models. It can model any pressurized water reactor. Its models include three-dimensional core neutronics, core thermal-hydraulics, a two-phase primary system, the steam generator secondary system, protective and relief systems, versatile control systems that can be made as simple or as intricate as desired, and a fully interactive user interface. CENTS also features a plant mimic with live parameter displays and live plotting capabilities, both of which are invaluable for plant response assessment during a transient simulation.

The first-principles models in CENTS provide excellent fidelity to actual plant behavior and reveal the detailed interactions between components. CENTS may be used to:

1. Assess the impact on overall plant performance and safety margin from control system interactions, design parameters, design features, design changes, and system interactions;
2. Provide contingency analysis results for potential events using optional recovery paths to mitigate the consequences of malfunctions and postulated accidents;
3. Aid in developing and assessing plant procedures and operating strategies;
4. Provide a training tool for reactor operators and other plant personnel;
5. Evaluate and validate existing full-scope simulators that use less accurate models;
6. Develop data for Emergency Response Drill scenarios;
7. Perform licensed safety analyses, pending approval by the U.S. Nuclear Regulatory Commission.

2.0 CODE DESCRIPTION

Background information, model descriptions, and the features and benefits of CENTS are provided in Sections 2.1, 2.2 and 2.3, respectively. A list of key code features is provided in Table 1. The features and capabilities of CENTS are products of C-E's experience in the development of design and simulation codes.

CENTS is an interactive, high-fidelity, user friendly, nuclear power plant simulation code. It models the plant as depicted in Figure 1, including the core (heat transfer and neutronics), primary system, secondary system through the turbine admission valves, and the associated safety, relief and control systems.

Use of first-principles models provides a high degree of fidelity and flexibility to the CENTS code. It allows CENTS to simulate a wide range of plant states from those of normal operation to multiple malfunctions, accidents and severe core uncover. Examples are loss of feedwater, natural circulation cooldown, steam generator tube rupture, steam line break and LOCA. Table 2 illustrates the range of transients that CENTS can simulate and gives further examples. The first principles models support a full range of interactive options between the user/analyst, the reactor control systems, and the NSSS. They fully support multiple failures and the effects of operator interventions or mistakes. In fact, nearly any user-defined event or combination of events that do not disrupt the core geometry can be simulated by CENTS.

2.1 Background

A full set of simulation software provides a fast and realistic simulation of any pressurized water reactor. CENTS is a modified version of the software that drives the full scope simulator developed by C-E for use by utilities. The reactor system simulation software is based on models taken from best-estimate design and licensing codes developed by C-E, and retains the verified high fidelity of these codes.

2.2 Models

The CENTS models are coded as a set of modules, each describing a particular portion of the nuclear power plant. An overview of the major models is given below.

Reactor Core Neutronics

The core neutronics module is an integrated group of programs: three-dimensional neutronics, point kinetics, core thermal-hydraulics and core damage. The 3-D program calculates the detailed power distribution of the core, with local power levels and fuel operating conditions for each fuel bundle. This provides a high level of accuracy to support simulated incore instrumentation. The point kinetics program uses the information on radial and axial power distribution to determine the net effect of localized events on reactor power.

The core neutronics module provides a realistic simulation of the three-dimensional core effects. To accomplish this in real time, three-dimensional and core-average calculations are performed for both neutronics and thermal hydraulics. Three-dimensional calculations are performed periodically to take into account spatial effects of control rods, delayed neutrons and thermal-hydraulic asymmetries. On the other hand, the core-average calculations, including point kinetics, are performed more frequently to provide smooth and stable simulation of core dynamic conditions and instrument response.

Three-Dimensional Neutronics. The 3-D neutronics model represents each assembly individually as a node, and further nodalizes each into four to ten axial sections. A typical neutronics nodalization at one axial level is illustrated in Figure 2. This nodalization is sufficiently fine to demonstrate the spatial and localized effects accurately. For example, there is never more than one control rod inserted in a node. Thus, each control or shutdown rod is simulated individually, along with its unique impact on the assembly neutron flux and power distributions. Complex questions of nodal cross section and coupling changes under various combinations of normal and abnormal rod insertions are thus avoided.

The model provides realistic responses for normal operations, and for all transients and malfunctions, including the localized effects of long term rod misalignment, rod motion either in banks or individually, rod drop, rod ejection and Xenon oscillation. It responds accurately to inherent feedback effects such as fission product poisoning, moderator temperature and Doppler feedbacks.

The neutronics code models the complete fission reaction including the production, transport and leakage of neutrons in the core. It simulates the spatial and time dependent effects of the startup and sustainer neutron sources, including photoneutrons produced from the dissociation of deuterium by fission product gamma particles. The model determines fission and decay heat release, thermal-hydraulic asymmetry effects, the detailed effects of grids and control rods on movable detector signals, and the spatial effects of burnup and refueling as seen at several times in a fuel cycle.

The core model can be easily updated for future core cycles, and is easily tunable for time-of-life.

Point Kinetics. The point kinetics model provides short-term temporal detail between the full three-dimensional calculations. It determines the instantaneous fission power by solving the standard point kinetics neutron equations with six delayed neutron groups, and calculates the decay power from an eleven fission-product group decay heat model. The decay power is based on the fission product inventory that corresponds to a specified cycle point.

The total reactivity in the point kinetics equation is calculated as the sum of feedback contributions from the moderator's temperature and boron content, the fuel temperature (Doppler feedback), normal movement of control rods, reactor scram, rod drop or rod ejection. The various reactivity contributions are updated at each calculational time step, based on the thermal, hydraulic and mechanical processes.

Core Thermal Hydraulics. The spatial detail for core thermal hydraulics (also for Xenon, Samarium and decay heat) is typically based on a 132-node model

consisting of 33 radial nodes at four axial levels. A typical arrangement is shown in Figure 3. Each core thermal hydraulic node represent between two and nine 3-D neutronic nodes, by applying a mapping technique to collapse the neutronic nodal powers.

Coolant flow and thermal processes are driven by the overall primary system model (below) and by the calculated rod heat fluxes. Energy transport in a node is determined by a closed channel formulation, except under low and reversed flow conditions when a mixing correlation helps to simulate the effect of buoyancy induced cross-flow.

Core heat transfer to the coolant is determined for both forced convection and quiescent pool boiling. A full range of heat transfer conditions is simulated including subcooled boiling, nucleate boiling, DNB, transition boiling, film boiling, and steam heat transfer. Fuel and cladding temperatures, heat flux, heat transfer regimes and surface heat transfer coefficients are calculated by means of an implicit numerical technique. This provides a very stable solution even for long time steps. The core heat transfer model describes the core for all modes of normal operation (e.g., cold shutdown to hot full power), mild transients and transients that result in significant core uncover and fuel cladding damage.

Fuel Failure. The fuel failure model predicts the extent of fuel damage, the subsequent fission product release and the rate of hydrogen generation. The fuel failure threshold is based on three design limits: (1) the hot channel DNB ratio, (2) the peak cladding temperature and (3) the amount of clad oxidation.

Departure from nucleate boiling (DNB) is usually localized and of short duration, resulting in an increase in coolant activity whose magnitude depends on the number of pins experiencing DNB. High cladding temperatures, which can occur during periods of core uncover, result in the oxidation of zirconium, generation of hydrogen and production of reaction heat, based on the Baker-Just correlation. Fuel pins that are oxidized beyond 17% or whose cladding temperature exceeds 2200°F are assumed to have failed. In addition, the amount of fuel damage as a result of DNB is defined as the product of the number of pins within a given pin power and the probability of DNB.

The fission product inventory available for release upon fuel failure is dependent on the power history. The model accounts for seven isotopes of iodine, xenon and cesium.

Primary Reactor Coolant System

The reactor coolant system (RCS) model describes the thermal-hydraulic behavior of the primary coolant system. The models are based on those developed for a best estimate version the CEFLASH-4AS¹ design code, which was developed for small break loss-of-coolant accident (LOCA) analyses.

¹CEFLASH-4AS is licensed by the NRC for C-E designed PWRs.

A typical node/flowpath geometric representation for the primary system of a four-loop, Westinghouse designed PWR is shown in Figure 4. Analogous representations are used for other plants having different configurations. The nodalization scheme is entirely generic, and may be redefined by the user via the CENTS database. Separate treatment of the loops provides appropriate response for partial and asymmetric loop operation, with separated counter-current flow represented in the hot and cold legs. Additional flow paths, not shown in the figure, provide for reactor vessel bypass, main pressurizer spray, and the addition and removal of coolant from the RCS.

The RCS model accounts for phase separation within each node into separate steam and liquid/two-phase regions. A two-phase region consists of a continuous liquid phase with dispersed bubbles produced by flashing, convection or heat sources. Phase separation is calculated by means of experimentally based correlations. The model accounts for the effects of fluid level on heat transfer, and on the quality and slip conditions of fluid in the flowpaths.

The RCS model provides a full range of nonequilibrium thermodynamic fluid states for all nodes. The model dynamically calculates each node's thermodynamic state, with separate two-phase mixture and steam regions, and interfacial exchanges for condensation and vaporization. The one-dimensional conservation equations (four for each node and one for each flow path) are integrated implicitly by means of a simultaneous solution of the linearized, discretized equations over all nodes and paths. Additional conservation equations couple these coolant conservation equations with ones for noncondensable gases, boron and fission products.

The pressurizer and upper head are each modeled as a single node with the same phase separation, wall heat and thermal nonequilibrium models as the rest of the primary system. Heaters, sprays, instruments and controls are provided in the pressurizer. Safety valves, power operated relief valves and manual vent valves are represented. The upper head modeling supports an accurate representation of upper head voiding and bubble management.

The CENTS code models all primary system components. The modeling includes consideration of reactor coolant pump (RCP) performance, wall heat transfer, pump heat, choked flow out of the primary system (through leaks or relief valves), and coolant thermodynamic properties.

The two-phase models for the reactor coolant system, pressurizer, and RCPs provide the capability to simulate pressurizer draining and refill, voiding of the reactor vessel upper head during events such as rapid natural circulation cooldown, degraded RCP performance, and significant RCS voiding (including core uncover) during simulated LOCA events.

Protective and Control Systems

CENTS Generic Control System. The CENTS Nuclear Plant Analyzer features a Generic Control System design that processes system parameters and produces signals to drive the various plant subsystems. The control system is modular in design. It is constructed by the system modeler to simulate his plant's control systems, and can be made as simple or as true to the actual controllers as desired. It has a complete inventory of functional modules, including arithmetic, Boolean, integro-differential and specialized functions. Once the

control system structure is established by the modeler, it functions automatically, and its details do not normally concern the CENTS user. However, the user can, at any time during a transient, interactively change setpoints, disable control systems or exercise manual control. The control system designer/analyst may study the detailed functioning of control modules by tracing their dynamic behavior, experimenting with their parameters and logic settings, or interfacing with the lines of communication among the control modules.

Reactor Protective System. The CENTS Nuclear Plant Analyzer models the Reactor Protective System (RPS). It activates a trip channel when a designated system parameter exceeds the corresponding trip setpoints. This system parameter may be a direct instrument reading or a processed combination of such readings, as defined within the CENTS Generic Control System (described above). Typical trip channels that have been defined include:

High power level	Low/high steam generator level
High power rate of change (startup)	Steam/feed flowrate mismatch
High pressurizer pressure	coincident with low level
Low pressurizer pressure	Low reactor coolant flow
Over-temperature thermal margin	High containment pressure
Over-power thermal margin	Scram on turbine trip
Low steam generator pressure	Scram on SIAS
	Manual scram

For any particular plant, only the trip channels appropriate to that plant are actually implemented and used.

Other Systems. CENTS also models actuation and operation of several subsystems including those found in the Engineered Safety Features Actuation System (ESFAS). An actuation signal is initiated when designated system parameters (direct or manipulated) exceed the corresponding setpoints. The following systems are programmed in CENTS: main steam isolation system (MSIS), emergency feedwater actuation system (EFAS) and safety injection system (SIS).

The SIS includes High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) based upon database supplied tables of flow versus back pressure. Safety Injection Tanks (SITs) or Accumulators are also modeled, including their elevation and pressure loss terms.

CENTS includes models of the major control and relief systems, including: reactor regulating system (including the control rod stepping mechanism), main and auxiliary feedwater systems, turbine bypass valves, steam dump valves, pressurizer pressure control system (proportional and backup heaters, main and auxiliary sprays), pressurizer level control system (letdown, charging, pressurizer level instrumentation), reactor power cutback function, and controls for automatic turbine setback and runback. The code also models the turbine admission valves, steamline isolation and check valves, and safety, relief and vent valves where appropriate in the primary and secondary systems.

Steam Generator Secondary Coolant System

The steam generator secondary system is represented by a nodal model, as shown in Figure 5 for a C-E designed PWR. Three nodes represent the secondary side of

each steam generator: a downcomer (subcooled or saturated), an evaporator tube bundle region (saturated or subcooled) and a steam dome (saturated or superheated). A fourth node, not shown in the figure, represents the economizer (preheater) region of steam generators with an economizer design.

The²steam generator secondary system model is based on models developed for the LTC² system design code. Mass and energy balances are made for each node. The circulation flow and the main steam flows are calculated using momentum balances. Steam separation from the evaporator to the steam dome is calculated using a bubble rise model. The pressure and remaining state properties are calculated from the mass and energy in each node.

Models for forward and reverse heat transfer are incorporated in the steam generator heat transfer logic. The overall heat transfer is determined from film resistance on the primary and secondary sides, and the resistance and heat capacitance of the tubes. The primary side film resistance for forward heat transfer is found from correlations for subcooled forced convection and two-phase condensation. The primary side reverse heat transfer is found from correlations for nucleate boiling in tubes and heat transfer to steam. The secondary side film resistance is calculated using pool boiling and heat transfer to steam correlations. Fluid level and its effect on tube heat transfer are modeled on both the primary and secondary sides of each steam generator.

This steam generator model provides the user with the ability to simulate a wide range of transients including: normal operational transients, excess load, MSIV closure, tube rupture, steam and feedwater line breaks on either side of the isolation and check valves, and once through cooling at off-design pressures and feedwater conditions. By modeling each steam generator separately, the model accurately predicts the asymmetric response of the system to transients initiated in one steam generator.

2.3 Feature and Benefits

CENTS databases are available for all classes of U.S.-designed PWRs. Each database provides a complete description of the nuclear plant systems modeled by CENTS for operation at full power in steady state. To initiate a transient from this steady state, the user effects appropriate changes or perturbations to the plant. Multiple plant states can be maintained simultaneously. A new plant state is obtained by running a simulation to interactively maneuver the plant from a given plant state, or by using the code's automatic self-initialization feature. Any intermediate state during a transient simulation can be saved for later study or to initiate parametric variations on plant behavior.

Use of the CENTS Nuclear Plant Analyzer is supported by a set of user friendly executive software that handles most of the simulation mechanics, and allows the user to interact with CENTS as the transient progresses. This software presents a graphical environment in which the user accesses popup and pulldown menus via a system of hotkeys, screen icons and mouse-driven control events. It also

²LTC is licensed by the NRC for C-E designed PWRs.

provides a sophisticated command language for the user who is keyboard oriented as well as for complex interactions and probes by the advanced user. The executive software supports changes desired in the course of the transient and facilitates evaluation of the plant behavior details. The user may freeze, resume or backtrack a transient simulation at any time, examine plant parameter, make changes, take manual actions and initiate malfunctions. The user may, at any time, instruct the code to automatically make changes, display parameters or take any interactive action at pre-selected times, or when pre-selected dynamic conditions are satisfied, if desired, without interrupting the simulation.

CENTS provides a number of output and display options. Its live system parameter plots and live graphic depictions of the plant state are invaluable tools in helping the user gain an understanding of the system behavior. The dynamic graphics may be saved for late study or demonstration in a playback mode. In addition, a combination of standard and user-defined numerical outputs allows the user to explore details of the plant subsystem behavior.

The CENTS Nuclear Plant Analyzer runs considerably faster than real time on its host computer workstation, with an optional real time mode. This allows the user to gain a sense of the real time behavior of the power plant.

3.0 ACQUISITION MODES AND SYSTEM REQUIREMENTS

The CENTS NPA may be acquired or accessed in one of several ways:

1. The CENTS NPA is available with a site-wide license, for installation on the following computer platform:

- o SUN workstation using the UNIX operating system and the SPARC central processor (i.e., the SUN 4/xxx, SPARCstation and SPARCserver families).

In addition, the following minimum computer specifications, peripherals and software are required:

- o 8 megabytes of main memory
- o Mass storage (hard disk) system with 300 megabytes capacity
- o Color monitor
- o UNIX operating system
- o FORTRAN and C compilers

Alternatively, a complete CENTS Station may be delivered, consisting of the above computer hardware with the CENTS NPA software installed on it.

2. The CENTS NPA may be installed on the user's workstation, minicomputer or mainframe computer, other than the one specified above. Details and system requirements depend on the computer system.

4.0 DOCUMENTATION

The CENTS User's Manual provides a detailed description of CENTS to facilitate use of the code. It contains a summary, a description of the models and a description of the inputs and outputs. It explains the core, primary, secondary, plant balance and control system models, to allow the user to understand CENTS and the results it produces. The documentation describes the database -- plant constants, setpoints and process variables -- providing definitions and full-power steady state values for each plant parameter. The manual also describes the inputs that are used to initiate and control transients. It discusses the important parameters that depict the plant state, with the aid of node-flowpath and component diagrams, to facilitate comprehension of the plant behavior during a transient simulation.

The Modeler's Manual for the CENTS Generic Control System provides complete information for accessing or building control system models, and for modifying, analyzing or tracing their functions.

The CENTS Training Manual offers a functional description of CENTS in introductory format, for the beginning user.

The CENTS Technical Manual offers instructions for installation and maintenance of the code.

TABLE 1

KEY FEATURES OF THE CENTS NUCLEAR PLANT ANALYZER

GENERAL

Desktop simulation with design code fidelity

Interactive or batch

User friendly

Minimum input due to use of Database

Faster than real time, with a real-time option

Multiple dynamic output options:

- Live graphical plant depiction

- Live parameter plots

- Numerical output (standard)

- Numerical output (user defined)

- Data files for post-simulation plotting

- Deep-probing user inquiries at any time

- Graphical playback

Backtrack and restarts

Self initialization for alternate system states

Fast-time formulation (2-60X) for plant evolutions

Table 1 (Continued)

CORE

Power

- Three-dimensional core neutronics
 - with
- Point kinetics with reactivity feedbacks
- Direct power deposition in cladding and moderator
- Decay power
- Zirconium-water reaction

Core Heat Transfer

- Fuel pin - axial/radial grid
- Complete boiling curves - forced and natural convection
- Implicit temperature and heat transfer solution

Special Component Models

- Core bypass
- Asymmetric core exit enthalpy distribution
- Two-phase drift flux:
 - Axial distribution of voids
 - Phase separation
- Fuel damage:
 - Clad oxidation
 - Hydrogen production
 - Fission product release

Table 1 (Continued)

PRIMARY COOLANT SYSTEM

Features

- Nodes and flowpaths
- One-dimensional, nonhomogeneous, nonequilibrium formulation
- Phase separation provides discrete level in nodes
- Slip flow between phases in flowpaths

Conservation Equations

- Five coolant equations
 - Mixture mass and liquid mass in each node
 - Mixture energy and steam energy in each node
 - Mixture momentum in each path between nodes
- Conservation equations for non-condensable gases, boron, fission products
- Implicit integration of coupled system - fast and stable
- Critical (choked) flow checks throughout the system
- Interfacial mass and energy exchange
- Local pressure and properties for each node
- Impact of non-condensable gases on local pressures

Special Component Models

- Pressurizer - see Control, Relief and Safety Systems
- Reactor Coolant Pumps - two-phase pump homologous curves, two-phase degradation, manual control, pump seal injection and leaks
- Wall heat - to fluid and containment
- Transport of fission products, boron and non-condensibles
- Accurate fluid properties from 0.1 to 6000 psia
- Quench tank / pressure relief tank

Table 1 (Continued)

SECONDARY COOLANT SYSTEM

Features

- Nodes and flowpaths
- One-dimensional, nonhomogeneous, nonequilibrium formulation
- Explicit solution of conservation equations
- Steam generator recirculation model

Special Component Models

- Downcomer
- Evaporator / tube bundle
- Economizer / preheater
- Steam dome
- Main steamlines and header
- Steamline check valves
- Steamline relief/safety valves
- Main feedwater pumps
- Fission product transport
- Steam generator blowdown system

Steam Generator Heat Transfer

- Level dependent
- Forward and reverse
- Boiling curves

Table 1 (Continued)

CONTROL, RELIEF AND SAFETY SYSTEMS

Features

- Control system - modular definition
- Individual control systems - automatic and manual modes
- Setpoints - default or user-selected values

Core

- Reactor Protection System
- Reactor (Control Rod) Regulating System
- Reactor Power Cutback Signal

Pressurizer

- Level Control System - Charging, letdown
- Pressure Control System - sprays (main & aux.), heater (control & backup)
- Relief System - PORVs, safety valves, vent, quench tank (PRT)

Safety Injection System

- Safety Injection Actuation Signal
- High and low pressure safety injection pumps
- Safety injection tanks / accumulators

Secondary Systems

- Main feedwater - level control, feed pumps, control valves
- Auxiliary feedwater
- Turbine admission valves
- Turbine bypass (steam dump) valves
- Main steam isolation valves and Isolation Signal
- Atmospheric steam dump valves
- Relief (safety) valves
- Turbine Cutback/Setback/Runback Signals

TABLE 2

EXAMPLES OF THE TRANSIENT CAPABILITIES OF CENTS

Steady State or Operational Evolutions: power change, heatup, cooldown

Increase in Heat Removal by Secondary System: stuck open relief valve,
feedwater malfunction, rupture of secondary system piping

Decrease in Heat Removal by Secondary System: loss or reduction of steam flow,
loss or reduction of feedwater

Decrease in Reactor Coolant Flow Rate: pump trips, shaft break or seizure

Reactivity or Power Anomalies: uncontrolled rod withdrawal, under-boration,
rod ejection

Increase in Reactor Coolant Inventory: ECCS actuation at power, charging or
letdown malfunction

Decrease in Reactor Coolant Inventory: inadvertent relief valve opening,
instrument line break, steam generator tube rupture, loss of coolant
accident

Anticipated Transients without Scram: loss of feedwater, inadvertent control rod
withdrawal, loss of steam flow

This Table is meant to indicate the versatility of CENTS. In fact, any user-defined event or combination of events, that involves plant components and systems modeled in CENTS, can be simulated.

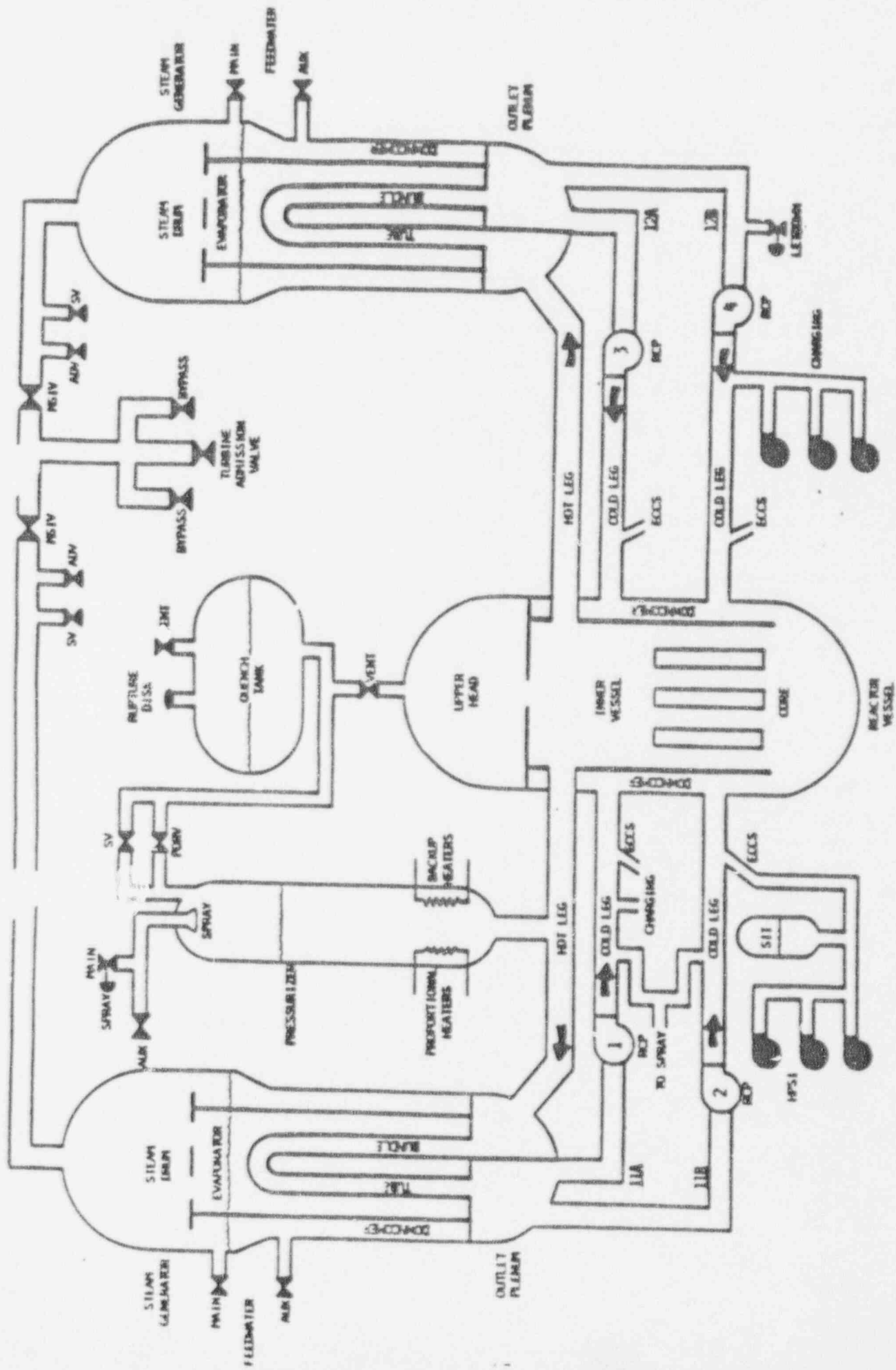
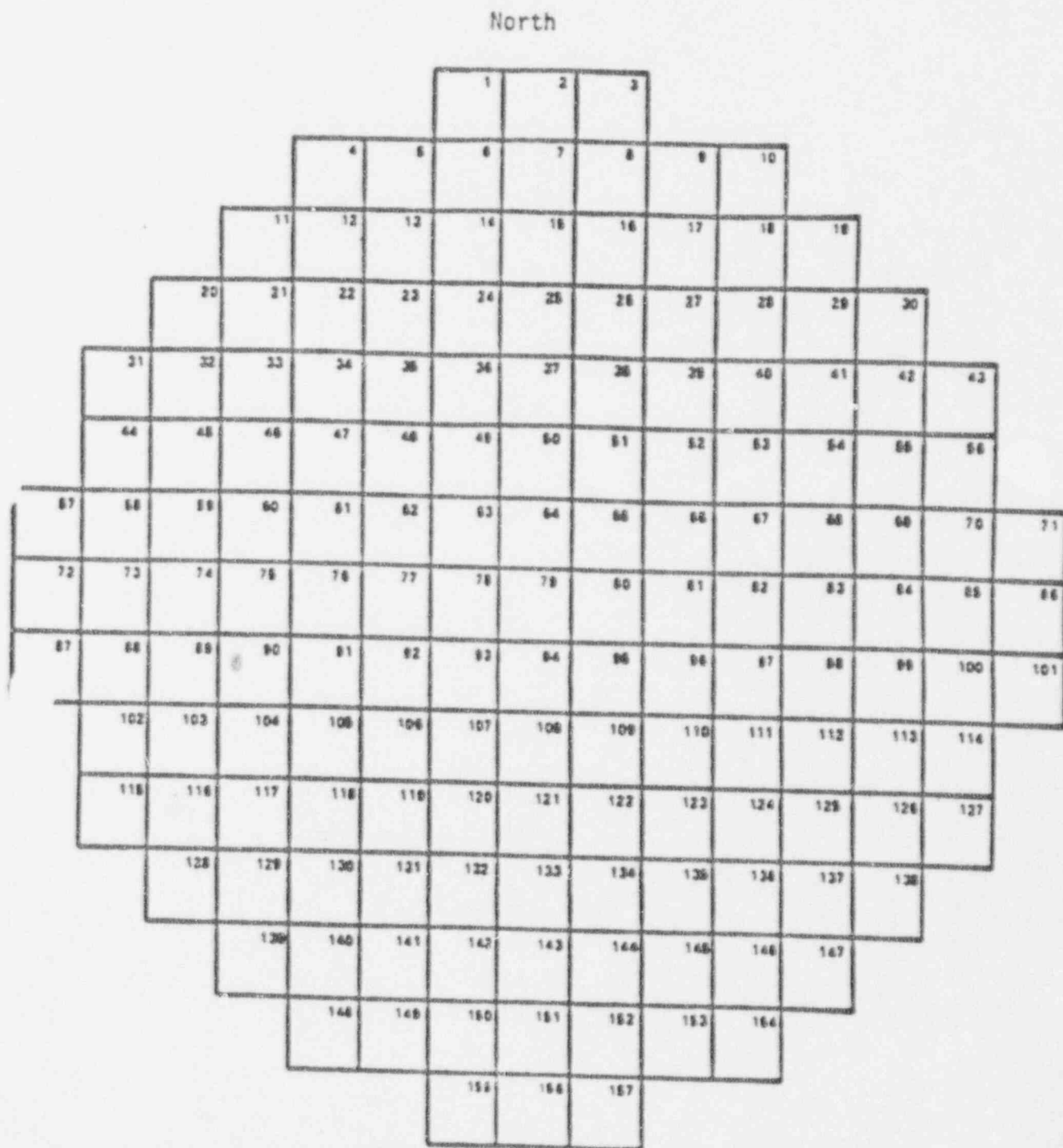


FIGURE 1
CENTS MODEL OF TWO-LOOP PRESSURIZED WATER REACTOR

FIGURE 2
THREE-DIMENSIONAL RADIAL NEUTRONIC NODES



NOTE: EACH FUEL ASSEMBLY IS EXPLICITLY MODELLED AT FOUR

FIGURE 3
THREE-DIMENSIONAL RADIAL THERMAL HYDRAULIC NODES

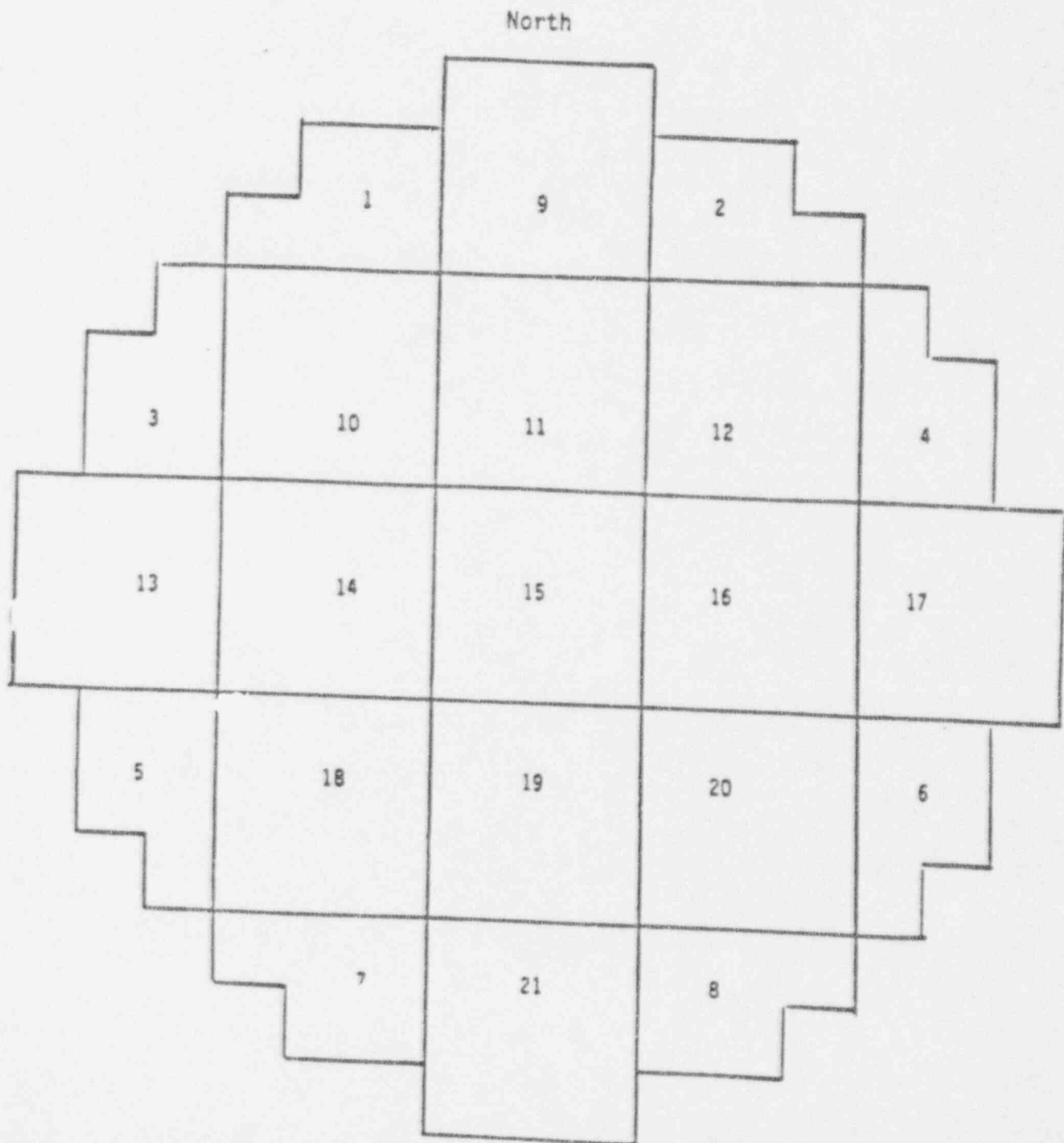


FIGURE 4

PRIMARY SYSTEM GEOMETRIC MODEL FOR A
4 LOOP WESTINGHOUSE PLANT

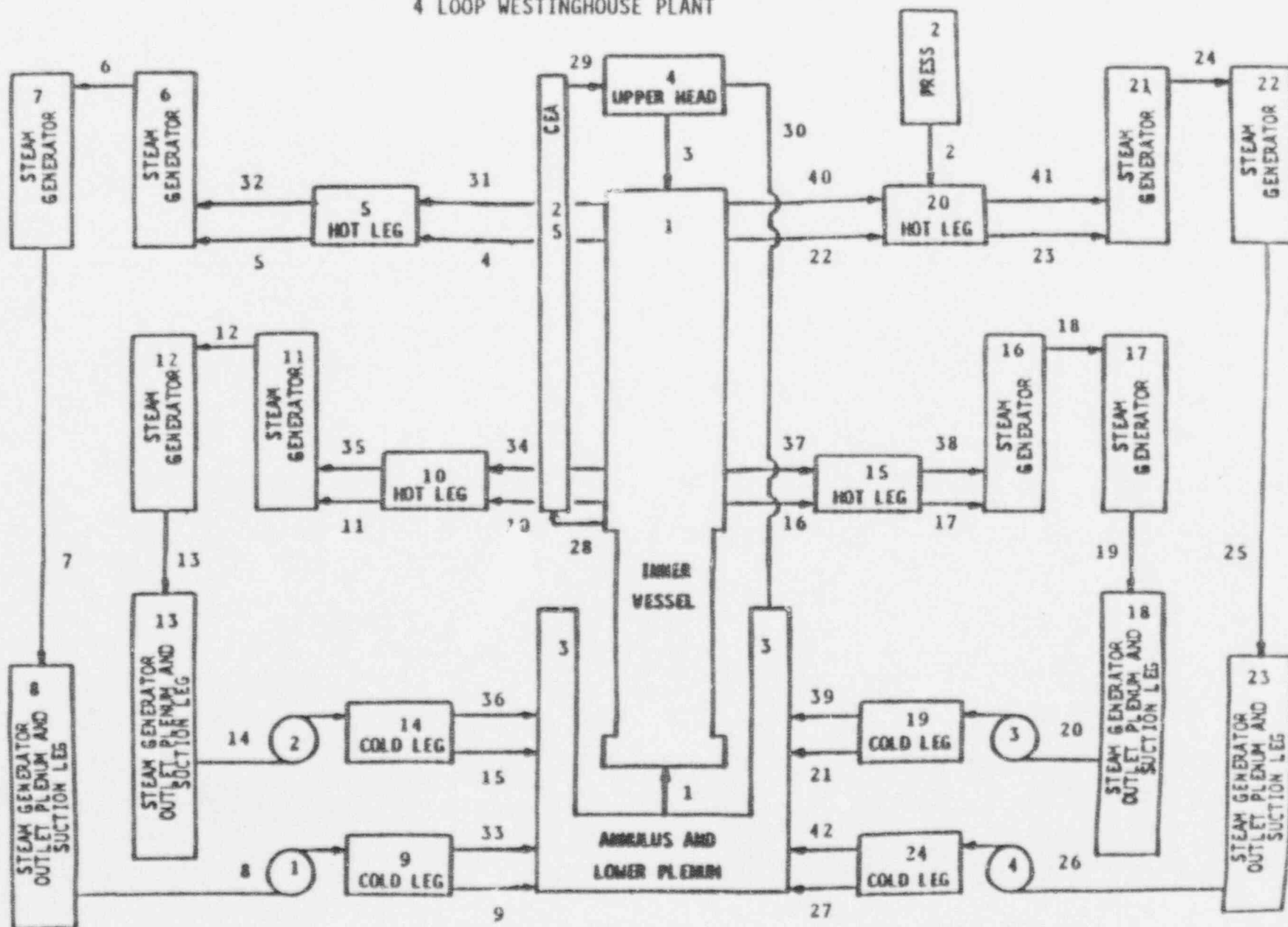
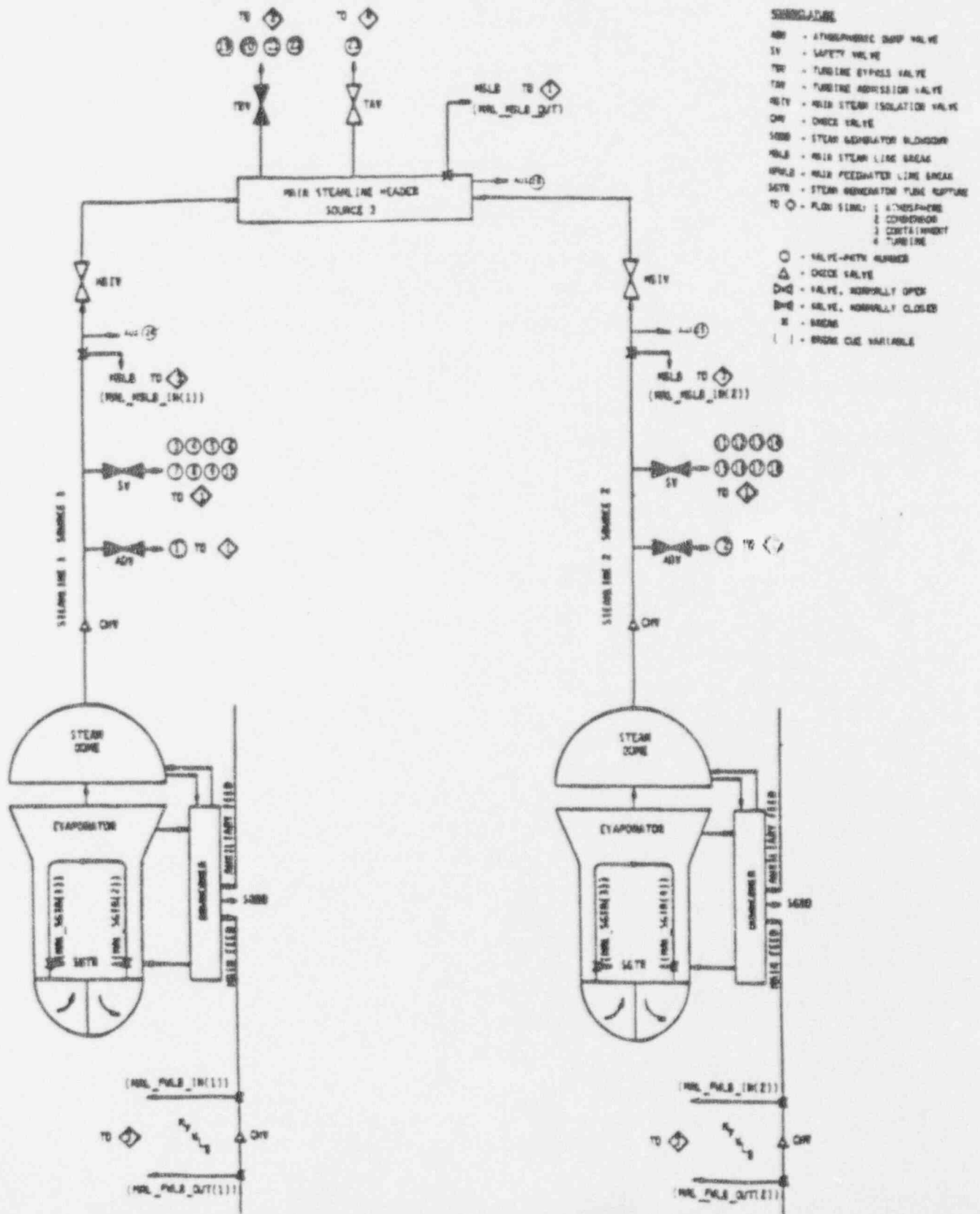


FIGURE 5

STEAM GENERATOR SECONDARY SYSTEM GEOMETRIC
MODEL FOR A TWO LOOP C-E PLANT



APPLICATION OF THERMAL-HYDRAULIC DESIGN
CODES TO REAL TIME PWR SIMULATION

by

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APPLICATION OF THERMAL-HYDRAULIC DESIGN CODES TO REAL TIME PWR SIMULATION

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SUMMARY

Improved thermal-hydraulic models for real time reactor simulation are described. They are based on models used in reactor design codes. The primary system model is formulated with five one-dimensional continuity equations conserving liquid and steam mass, liquid and steam energy and momentum. The models incorporate two-phase, nonhomogeneous, nonequilibrium capabilities based on first principle physics. Validation of the design codes by reactor test data is described. The ability of the models to predict proper system behavior is demonstrated for a pump transient, an overcooling transient, and a small break loss of coolant accident (LOCA).

The feasibility of developing and integrating models for the primary and secondary system which execute faster than real time on computers used in simulators is shown. The resulting simulator thermal-hydraulic models can directly calculate plant response for a wide range of normal and abnormal conditions including LOCAs. Full provision for operator intervention or multiple failure interactions is available. Reality of plant behavior is maintained and the capability to perform calculations in support of design studies and development of operating procedures is provided.

INTRODUCTION

Presentations given at the conference on Simulation Methods for Nuclear Power Systems held two years ago highlight limitations of the power reactor simulators in use at that time¹. The models are limited to a single-phase fluid representation. Phase-separation and nonequilibrium effects are not represented. As a consequence, the more severe transient responses are often preprogrammed. Operator interaction with preprogrammed transients is not possible. Unrealistic responses are obtained in many cases with a possibility of resultant inefficient training of the operator.

Since the Three Mile Island Unit 2 (TMI) accident in March 1979, new capabilities have been defined for power reactor simulators. The ANS 3.5 Standard lists a number of the required capabilities including normal plant evolutions and abnormal conditions resulting from malfunctions². The post-TMI NRC Action Plan lists additional requirements particularly in the Reactor Coolant System (RCS) Models³. Among the currently required capabilities are the following. Operator interaction with the transient is needed where operator actions can influence the severity of the malfunction. The simulation must continue until a stable, safe and controllable condition is attained which can be continued to cold shutdown conditions. Observable control room indications must be realistic to provide proper training of the operator. Multiple failures or malfunctions must be modeled with appropriate interactions. An additional desired feature is the ability to do engineering calculations to support design studies and development of operating procedures.

Satisfaction of these requirements demands addition of new first-principle physics models. Most important is the addition of two-phase modeling capability for saturation conditions which occur in natural circulation cooldown as well as the more severe transients such as a small break LOCA. The stuck open pilot operated relief valve (PORV) at TMI produced a small break LOCA transient. Such a transient requires phase separation capability and nonequilibrium models to provide realistic pressure transients when fluids of very different temperatures interact. Real time simulation is required, so very efficient algorithms are needed.

Existing thermal-hydraulic system design and accident analysis codes provide the desired physical models. They are routinely used to analyze a full range of thermal-hydraulic transients including accidents leading to two-phase fluid conditions. However, these codes have been of limited use in training simulators since the computers used in simulators cannot provide the required computational speed to maintain real time simulation. The design codes also lack the detailed representation of system hardware required to drive the simulator control panel. The problem then is to combine the more complete physical models of the design codes with the real time requirements of a simulator.

COMBUSTION ENGINEERING SIMULATOR

Combustion Engineering is currently building a full scope control room training simulator for a Pressurized Water Reactor (PWR). The simulator models the complete PWR -- core, primary system, secondary system, turbo-generator, balance of plant, control systems, control panel, plant computer and instrumentation. It is capable of simulating plant behavior in real time during normal and abnormal operation. It simulates the dynamic response of the power plant to operator actions and system malfunctions activated by the instructor. It is designed to be utilized as a training device, as a tool for development and verification of new operating procedures, and as a tool for plant improvement studies.

Modeling of the Nuclear Steam Supply System (NSSS) is based on state-of-the-art design codes used for design and analysis of the major NSSS components and systems. The design codes have been verified with respect to PWR plant response data, to integral test facilities such as the Loss of Fluid Test (LOFT) facility or to separate effects tests. Additional models are provided for balance of plant systems.

The simulator uses computer hardware with proven performance and reliability. A Perkin-Elmer model 3244 computer in a four computer multiprocessor, shared memory configuration is used. This supplies the required computational capability with provision for support of related computational needs. Final code is in FORTRAN 77 compatible with the ANSI Standard. A unified software package manages the computer resources, includes generalized solution packages for balance of plant systems, and provides consistent software standards for coding of models.

The control panels closely parallel the panels of the reference plant. They include the full range of control, indicating and annunciating devices found in the power plant. An instructor's panel is also included to run the simulator and activate malfunctions during operator training. Over 200 malfunctions are provided for the instructor.

THERMAL-HYDRAULIC MODEL DESCRIPTION

The C-E Full Scope Simulator is designed to provide realistic thermal-hydraulic responses and instrument indications for a full range of reactor operating conditions. The models are based on those found in computer codes used for PWR design. The primary system or Reactor Coolant System (RCS) models are based on those used in a best estimate, design version of the CEFLASH-4AS computer code⁴. This code provides realistic RCS responses for a full range of system conditions including a loss of coolant accident. The secondary system models are taken from the Long Term Cooling (LTC) computer code⁵. The code is used to evaluate the integrated plant response to operational and accident conditions. Highlights of the more important models are given here.

Primary System Geometric Representation

The primary system thermal-hydraulic response is modeled by a node and flowpath network. The nodes enclose control volumes which represent the fluid mass and energy. Flowpaths connecting the nodes represent the fluid momentum and have no volume. The separation of mass and energy into control volumes and momentum into flowpaths is similar in concept to that used in the FLASH-4 code⁶.

The simulator geometric representation for a typical C-E PWR (for all conditions but a large break LOCA) is shown in Figure 1. Nodes are provided for the major components in the reactor primary system — inner vessel, upper head, control element assembly guidetubes, two hot legs (including the associated steam generator inlet plenum), a pressurizer, two steam generators

(separate hot and cold sides of the U-tubes), two combined outlet plenum and suction legs, four cold legs and reactor coolant pumps, and the annulus or downcomer for the reactor vessel. A total of 17 control volume nodes are used in the RCS model. The separate cold leg representations provide appropriate responses for partial loop operation. The interface with the secondary system occurs at the steam generator tube bundle.

More than 20 flowpaths with momentum solutions are used in the RCS model. Separated counter-current flow is represented in the hot and cold legs. An additional 32 non-momentum flow paths are provided for addition and removal of coolant from the RCS by the emergency core cooling system (ECCS), pressurizer spray, charging pumps, letdown system, shutdown cooling system, and safety and relief valves.

The noding and flowpath modeling detail is not hardwired. The detail in representation of the system can be changed through input during the modeling process to tailor the representation to the requirements of a particular PWR design. Geometric features such as the number of loops or the number of steam generators as well as geometric details such as node volume, elevations, piping resistance, etcetera, are specified in a data base. Adoption of the RCS models for another plant does not require extensive recoding of the software as is done in earlier simulators.

Conservation Equations

The RCS thermal-hydraulic model is formulated with five one-dimensional continuity equations. The conservation variables are mixture (liquid and steam) mass, liquid mass, mixture energy, steam energy, and mixture momentum. The mass and energy for the liquid and steam are calculated for each node. Mass flowrate is calculated for each flowpath. The code incorporates slip effects in the flowpaths by means of empirical correlations.

The conservation equations are integrated implicitly by means of a simultaneous solution of the linearized, discretized conservation equations. This yields a $(4 \times M + N)$ by $(4 \times M + N)$ system of linear equations, where M is the number of nodes and N is the number of momentum flowpaths. The structure of the coefficient matrix permits use of a system reduction procedure which produces a $N \times N$ system of equations. The $N \times N$ matrix is solved using a block inversion technique.

After solution of the conservation equations, the pressure in each node is calculated. The mass and energy in the liquid and steam regions are used with the water property correlations and nodal volume to determine the resulting state variables (pressure, phase enthalpies and phase temperatures).

Two-Phase Fluid Representation

The simulator models phase separation within a node into a separate steam region and a liquid or two-phase region consisting of a continuous liquid phase with dispersed bubbles. Bubbles are generated by heat sources within the

node, flashing in the node or transport from the adjacent nodes. Phase separation is calculated in terms of bubble rise velocities which are found from experimentally based drift flux correlations⁹. A more detailed model provides an axial bubble mass distribution in the inner vessel node¹⁰. Nodes with phase separation provide a discrete two-phase mixture level in the node. Fluid level effects on heat transfer and the quality of fluid exiting through flowpaths connected to the node are modeled.

Nonequilibrium States

The simulator provides a full range of thermodynamic fluid states for all nodes. Nodes with homogeneous or fully mixed fluid are at equilibrium. States for nonhomogeneous or phase separated nodes with separate two-phase mixture and steam regions are subcooled liquid-saturated steam, saturated liquid-superheated steam, and subcooled liquid-superheated steam. The model dynamically calculates the node thermodynamic state. It includes a detailed flow regime dependent condensation model which considers condensation of bubbles, vapor condensation on an injected subcooled liquid, condensation at the surface of a liquid pool, and vaporization due to wall heat. In addition, energy partitioning of wall heat transfer between the liquid (two-phase) and steam regions is calculated.

Reactor Core

Reactor power is calculated, by a three-dimensional reactor core model described in these proceedings¹¹. An axial power distribution and radial power tilt are taken from the reactor core model. Axially varying heat transfer and core coolant data is provided to the core model to provide a full range of temperature and moderator driven feedbacks.

The core heat transfer model covers the full range of fluid conditions in the core. Provision is made for forced convection and quiescent pool boiling conditions. The mode of heat transfer is determined dynamically. Boiling curves for both conditions include subcooled, nucleate boiling, transition boiling, film boiling, and steam heat transfer correlations. Fuel temperature, cladding temperature, and the heat flux, hence, the heat transfer regime and surface heat transfer coefficients, are calculated implicitly. Appropriate radial noding detail is used in the fuel rods. The axial nodal detail is varied depending on whether forced convection or pool boiling is occurring.

Other Primary System Models

The simulator models all primary system components. Some additional models of interest are discussed here. Reactor coolant pump (RCP) performance is found by conservation of angular momentum and homologous curves for single- and two-phase conditions with two-phase head degradation. Wall heat transfer, pressurizer heaters, and pump heat are modeled. Coolant inventory changes due to systems such as charging pumps, the emergency core cooling system, the chemical volume control system, and auxiliary pressurizer spray are calculated using a general purpose, single-phase flow solution software package. Critical

flow out of the primary system, through leaks or relief valves, is found from standard critical flow correlations¹². Generation of non-condensable gases and their collection in the upper head are modeled. Coolant thermodynamic properties are found from a set of fast water property correlations specially designed to provide continuous property derivatives for a full range of fluid conditions — subcooled, saturated, and superheated.

Steam Generator Heat Transfer

Models for forward and reverse heat transfer are incorporated in the steam generator heat transfer logic. The overall heat transfer is determined from film resistance of the primary and secondary sides and the wall resistance. The primary side film resistance for forward heat transfer is found from correlations for subcooled forced convection and two-phase condensation. The primary side reverse heat transfer is found from correlations for nucleate boiling and heat transfer to steam. The secondary side film resistance is calculated using pool boiling and heat transfer to steam correlations. Fluid level is modeled on both the primary and secondary side of each steam generator. The effect of fluid level on heat transfer area is modeled on both sides.

Steam Generator Secondary System Geometric Modeling

For a plant with two steam generators, the steam generator secondary system is represented by a seven node model, Figure 2. Three nodes are used for the secondary side of each steam generator — a downcomer (saturated or subcooled), an evaporator region (saturated or subcooled), and a steam drum (saturated or superheated). One additional node represents the common steam line header. This system representation allows accurate modeling of the recirculation phenomena and the downcomer and evaporator water levels. In addition, by collapsing nodes, the model can represent dry steam generators. All major components are modeled, including the secondary safety valves, atmospheric dump valves, and main steam isolation valves. All main steam flow paths are considered.

Secondary System Conservation Equations

Mass and energy balances are made for each node. The flow between the downcomer and the evaporator, and the main steam flows are calculated using momentum balances. The recirculation flow from the evaporator to the steam drum is calculated using a bubble rise model. The pressure and remaining state properties are calculated from the mass and energy in each node.

REAL TIME CAPABILITIES OF THERMAL-HYDRAULIC MODELS

An important consideration in the selection of the thermal-hydraulic models for the simulator was the ability to obtain a faster than real time calculational speed. The design code thermal-hydraulic models selected as a basis for the primary and secondary systems generally run much faster than

real time on a high speed computer such as the CDC 7600. However, the Perkin-Elmer 3244 computer used in the C-E simulator would not provide the needed calculational speed for the design code RCS models. Two classes of changes were made to the primary system design code models to satisfy the real time computational requirement -- model optimization and model changes allowing use of long and constant time steps. Code optimization provided substantial time savings. An example is the water property package. Use of high speed polynomial fits, less frequent evaluation of properties which change slowly, and linearization of properties reduced the computational time to one-tenth that required formerly. Also, simplification of design code models reduced computational time. Removal of options and detailed models particular to design calculations, which are not required for the simulator, provided the needed simplification.

Use of a constant time step length allows the simulator thermal-hydraulic models to keep up with real time. It requires accommodation of potential discontinuous transitions which would produce abrupt changes observable by the operator or even instability with constant time steps. An example is addition, in a single time step, of more liquid to a node than the available steam volume can hold. This overfilling, or node packing phenomenon, produces a severe pressure spike. Use of an approximate solution technique provides smooth pressure results until normal conditions are obtained. A similar technique applies to overdraining a node, removing more water in a time step than was present. Linearization of other equations describing transitions also allows elimination of discontinuous behavior. Use of more implicit mathematical representations was also needed for some models. Fully implicit modeling of the core heat transfer, bubble release calculations, and pump speed calculations allows use of much longer time step lengths. Use of larger nodes also reduces calculational time by increasing the permissible time step length. Some temporal detail is lost with little effect on the information seen by the operator. The model performance is fully satisfactory.

Less geometric detail in the primary system noding and flowpaths is used for the large break LOCA calculation. The RCS time step length is reduced. No change is made in the secondary side model. This allows direct calculation of a large break LOCA in real time.

The computational speed of the thermal-hydraulic models has been tested by running a wide range of operational simulations from steady state through accidents with two-phase fluid conditions. The computational speed results are shown in Table 1. The combined time for the primary and secondary thermal-hydraulic models is much faster than real time, providing the desired overall real time capability for the simulator. The same constant time step is used for both the primary and secondary system models.

Table 1

Computational Time for Thermal-Hydraulic Models*

	<u>Primary</u>	<u>Secondary</u>
Perkin-Elmer 3244	250	40
CDC 7600	20	---

*Computational time in milliseconds per second of real time.

MODEL VALIDATION

Validation of the design codes upon which the simulator models are based is obtained, in part, by comparison of analyses of experimental transients and data from experiments. The best estimate version of the CEFLASH-4AS code has been extensively tested by such comparisons for both separate effects tests and integral tests on both the Semiscale and LOFT facilities. Two integral system test comparisons are shown for CEFLASH-4AS. The design code used as a basis for the secondary system models has been tested with separate effects data and with power plant performance data. One power plant transient is shown for this code.

LOFT test L3-1 simulated a 0.09 square foot single ended pump discharge break, a small break LOCA, for a full size PWR. A comparison of the system behavior calculated with the best estimate CEFLASH-4AS code and experimental results is shown in Figure 3¹³. The agreement between the data and the prediction is excellent. The predicted time of accumulator actuation is almost exact and the predicted behavior of the reactor vessel two-phase mixture height before and after accumulator actuation is very similar to the data. Predicted accumulator discharge does not result in a rapid depressurization and a high accumulator flow rate as normally occurs with thermal equilibrium codes.

Confirmation of the ability of the CEFLASH-4AS code to represent reactor behavior for a two-phase transient with the reactor coolant pumps (RCP) running is provided by the LOFT L3-6 test¹⁴. This test simulated a 0.1 square foot cold leg small break LOCA with the RCP powered throughout the transient. Figures 4 and 5 compare best estimate CEFLASH-4AS calculations with the test results. The analysis predicts the primary pressure, Figure 4, quite accurately. Reasonable agreement is provided for the primary system inventory, Figure 5. The homogeneous and nonhomogeneous or separated flow regimes agree very well with gamma densitometer data from the test. These results and those for the LOFT L3-1 test demonstrate the ability of the CEFLASH-4AS best estimate code to accurately model reactor thermal-hydraulic behavior during severe transients such as a small break LOCA. The next section demonstrates a comparable calculational capability for the simulator.

Comparisons of nuclear plant calculations made by the code upon which the secondary system models are based, with experimental results, are presented in Reference 15. Power plant test data including step changes in power, plant trips, and reactor coolant pump trips are discussed. A plant cooldown transient under natural circulation conditions and an overcooling transient are included. The design code is shown to be valid for a variety of events of interest. A high degree of correlation between the analytical predictions and the test data is shown to exist. Results for the overcooling transient, found with the simulator model, are presented in the next section.

PERFORMANCE OF SIMULATOR THERMAL-HYDRAULIC MODELS

The simulator thermal-hydraulic model performance has been evaluated for a wide range of PWR operation from normal plant evolutions to severe accidents. The accuracy of the steady state predictions is confirmed by their consistency, comparison to plant data, and comparison to design code calculations. The steady state calculations demonstrate correct implementation, proper interfacing, and numerical stability of the models. Operational transients are matched to plant performance or design code calculations. These include power changes and startup or shutdown of various devices including the RCPs, pressurizer sprays or heaters, and inventory control systems such as the ECCS systems. Many of the abnormal or accident transients normally can only be compared to design code analyses. Representative accident transients are small and large break LOCAs, steam-line breaks, losses of feedwater, and anticipated transients without scram. Appropriate behavior for these transients is necessary to provide optimal operator training.

Three examples of the testing program are shown here. The first is a single-phase RCP transient including steady state conditions and a series of abrupt changes in RCP operation. Next is an overcooling transient. Last, is a small break LOCA including an extended period with two-phase conditions, core uncover, and initiation of core recovery. The test cases were run with the simulator thermal-hydraulics software on the Perkin-Elmer 3244 computer. All cases were run with real time computational speeds similar to those shown in Table 1.

Single-Phase Reactor Coolant Pump Transient

The RCP transient is designed to show proper steady state operation and to demonstrate the thermal-hydraulic model's response to abrupt change in single-phase coolant flow. It begins at full power. Next the core is scrammed and the RCPs and turbine are tripped. This is followed by restart of each RCP singly, and shutdown of two pumps. The transient is summarized in Table 2.

Table 2
Reactor Coolant Pump Transient Event Summary

<u>Time (Seconds)</u>	<u>Event</u>
0	Start steady state at full power
50	Scram reactor, trip all RCPs, trip turbine.
100	Restart Pump 1*
150	Restart Pump 2
200	Restart Pump 3
250	Restart Pump 4
300	Trip Pumps 2 and 3
350	End of transient

*Pump numbers are shown on Figure 1.

Figures 6 and 7 show the pressurizer pressure and level. Figures 8-11 show the flow rate through reactor coolant pumps 1-4, as designated in Figure 1. The system maintains a constant steady state for the first 50 seconds. No drift or oscillation is observed. Reactor scram and pump trip at 50 seconds is followed by pump coastdown. Pressurizer level drops throughout the transient due to inventory shrinkage by the coolant density rise. Charging pumps are not actuated as the pressurizer level drops.

The effect of restarting one pump every 50 seconds is shown in Figures 8-11. Coolant flow through each restarted pump rises abruptly. When pump 1 is restarted, flow through the tripped pumps rapidly becomes negative. As each additional pump is restarted, the flow through the previously operating pumps drops and the magnitude of the negative flow in the tripped pumps increases as expected. Trip of pumps 2 and 3 at 300 seconds rapidly produces the same flow rate in pumps 2 and 3, a negative flow, and in pumps 1 and 4, a positive flow. The path of the coolant flow can be traced in detail by referring to Figure 1. The coolant flows from pump 1 through the cold leg to the annulus. Part of the coolant goes through the core, hot leg and steam generator back to pump 1. The remainder goes back through the adjacent cold leg in reverse flow through pump 2 to the steam generator outlet plenum, and finally through pump 1 in forward flow. The same process is repeated for the components on the other side of the reactor vessel. The ability of the models to represent both symmetric and asymmetric flow distributions with both forward and reverse flow through the RCPs is demonstrated.

Overcooling Transient

A steam generator secondary system overcooling transient is used to illustrate the ability of the simulator thermal-hydraulic models to match measured PWR data. The pressurizer level and pressure, steam generator secondary pressure and level, and the hot and cold leg temperatures calculated by the simulator are compared to measured plant data¹⁵. The simulator calculation is done for a plant with a lower power level than the reference data plant. Some difference in plant behavior is expected due to differences in the plants, but a meaningful comparison is expected because the parameters controlling the transient were selected to produce the same key events and similar trends in plant behavior. The overcooling transient is initiated by a turbine trip from full power after which a steam bypass valve fails to close. The initiating events are summarized in Table 3. This is the same PWR transient discussed earlier in connection with validation of the steam generator secondary design code.

Table 3
Overcooling Transient Initiating Event Sequence

<u>Time</u>	<u>Event</u>
0	Turbine trip (manual)
1	Steam dump and bypass valves begin to open
6	Reactor trip (on low steam generator level)
18	Emergency feedwater to steam generators
21	One steam bypass valve fails to close
200	Reactor coolant pumps tripped (manual - on low primary pressure)
240	Main steam isolation valves closed (on low secondary pressure)
240	Main feedwater isolation valves closed
1200	Plant conditions stabilized

Following transient initiation by manual turbine trip the steam generator secondary pressure, Figure 12, rose rapidly until scram. Primary pressure also rose rapidly until scram, Figure 13. Steam generator level dropped after scram, Figure 14*. Pressurizer level rose initially until scram after which it fell, Figure 15. The simulation held one steam generator bypass valve open to

*Steam generator level is referenced to the level of the narrow range pressure tap.

represent the failure experienced in the PWR transient at 21 seconds. This initiated an overcooling event on the steam generator secondary side which caused the pressure and level to drop rapidly, Figures 12 and 14. The overcooling also caused the pressurizer pressure and level to drop rapidly, Figures 13 and 15. This portion of the transient provides a significant challenge to the simulator model's ability to calculate the compression and expansion experienced in the tests. The simulator reproduces the system behavior shown in the PWR test data.

Closure of the main steam isolation valves at 240 seconds ended the plant cooldown. After this, addition of emergency feedwater caused the steam generator secondary level and pressure to rise gradually. The rise in secondary temperature produced an increase in pressurizer pressure and level. Again the simulator reproduces the PWR transient during the pressure recovery. The thermal-hydraulic models successfully predict the significant events of the steam generator secondary pressure and water level dynamics. The nonequilibrium models and detailed momentum solution provide an appropriate prediction of the pressurizer pressure. The simulator successfully predicts the pressurizer water level indicating an adequate prediction of the RCS temperature history. Figure 16 shows this successful RCS temperature prediction capability for the hot leg and cold leg temperatures. Establishment of natural circulation and the resulting increase in core coolant temperature rise, after the RCPs are tripped, is demonstrated.

Small Break Loss of Coolant Accident

A small break LOCA transient is used to illustrate the simulator thermal-hydraulic model's ability to handle a severe accident. This transient is particularly challenging since the two-phase fluid conditions, core uncover, core recovery, and emergency core cooling system cold water injection test the phase separation, nonhomogeneous and nonequilibrium models. The simulator analysis results are compared to those from a best estimate CEFLASH-4AS calculation. The significant initiating events for the transient are described in Table 4.

Table 4

Small Break LOCA Initiating Event Sequence

<u>Time (Seconds)</u>	<u>Event</u>
0	Open 0.12 ft ² break in one cold leg
11	Scram core, trip turbine, trip RCP
50	Initiate HPSI
2500	End of transient

The small break LOCA is initiated by opening a 0.12 square foot break in one cold leg of the primary system. The major events and trends of the transient are summarized in Table 5. The best estimate and simulator primary pressure predictions are shown in Figure 17. The pressure drops rapidly for 80 seconds, plateaus with a moderate rise until 270 seconds, and drops gradually until 1500 seconds when it levels off. The inner vessel two-phase mixture level prediction by the best estimate code and the simulator are shown in Figure 18. The level drops rapidly until 70 seconds, pauses until 400 seconds, and falls gradually due to coolant boil-off until about 1200 seconds when core recovery begins. Core uncover occurs at about 700 seconds. The core exit fluid temperature predicted by the simulator is shown in Figure 19 with the saturation temperature. The temperature is subcooled initially, stays at saturation until the core uncovers, rises to a peak and falls toward saturation shortly after core recovery begins.

Table 5

Small Break Transient History

<u>Time (Seconds)</u>	<u>Discussion</u>
0	LOCA initiated by opening 0.12 ft ² cold leg break, rapid drop of inner vessel pressure and two-phase mixture level
20	Transition from subcooled to saturated blowdown
40	Pressurizer drained
70	Pause in inner vessel level change. Upper head and plenum drained, draining of steam generators and hot legs begins
80	Plateau in primary pressure just above secondary heat sink pressure
270	Pressure drop resumes upon transition to steam blowdown
400	Inner vessel level resumes gradual decline after draining higher elevation components
700	Core uncover begins, temperature of core exit steam rises above saturation
1200	Minimum core level reached, refill begins
1500	Pressure stabilizes at 250 psia
1550	Core exit steam temperature peaks and soon begins a gradual decline

The simulator reproduces the small break LOCA behavior predicted by the best estimate design code CEFLASH-4AS. The trends and events of Table 5 which would be observed by the reactor operator are reproduced with similar timings. These include the initial abrupt pressurizer pressure and level drop, the pressure plateau, the departure of core exit fluid temperature from saturation to superheat conditions, the pressure stabilization, and the core exit fluid temperature peak followed by a decline. Comparison of the predicted primary pressures in Figure 17 shows good agreement in the details of the pressure history. The inner vessel level predicted by the simulator lags slightly behind that of the best estimate code, Figure 18, but shows the same overall behavior. The lag is due to simplifications in the simulator models. The good agreement is the result of the two-phase and nonequilibrium models which give the simulator the ability to realistically predict reactor behavior for severe accidents.

CONCLUSIONS

The C-E simulator thermal-hydraulic models are validated for a variety of events which must be represented by a simulator. The ability of the models to hold a steady state and handle symmetric and asymmetric single-phase pump transients is shown. The capability to predict measured PWR performance data for an overcooling transient involving nonequilibrium and two-phase conditions is demonstrated. The capacity of the models to reproduce the reactor behavior for severe transients, such as a small break LOCA, which is found with a best estimate design code is shown. The feasibility of developing and integrating models for the core, RCS and steam generator secondary systems which are based on design codes and which execute in real time is proven. The capability of the two-phase nonhomogeneous, nonequilibrium models is shown.

Use of models based on first-principle physics has extended the capability of the C-E simulator beyond that shown by earlier simulators. It can directly calculate a wide range of events—steady state, operational transients, and abnormal conditions such as total loss of feedwater, steam line break, steam generator tube rupture, anticipated transient without steam, and small and large break LOCAs. Full provision for operator intervention to change the course of the transient is available. Multiple failures and their interactions are permitted. The models are generic to a wide range of PWR plants with geometric model changes including noding accomplished through input data. Reality in the transient behavior is maintained to provide optimal operator training. Engineering calculations to support design studies and procedure development are possible.

ACKNOWLEDGEMENTS

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FIGURE 1
PRIMARY SYSTEM GEOMETRIC MODEL

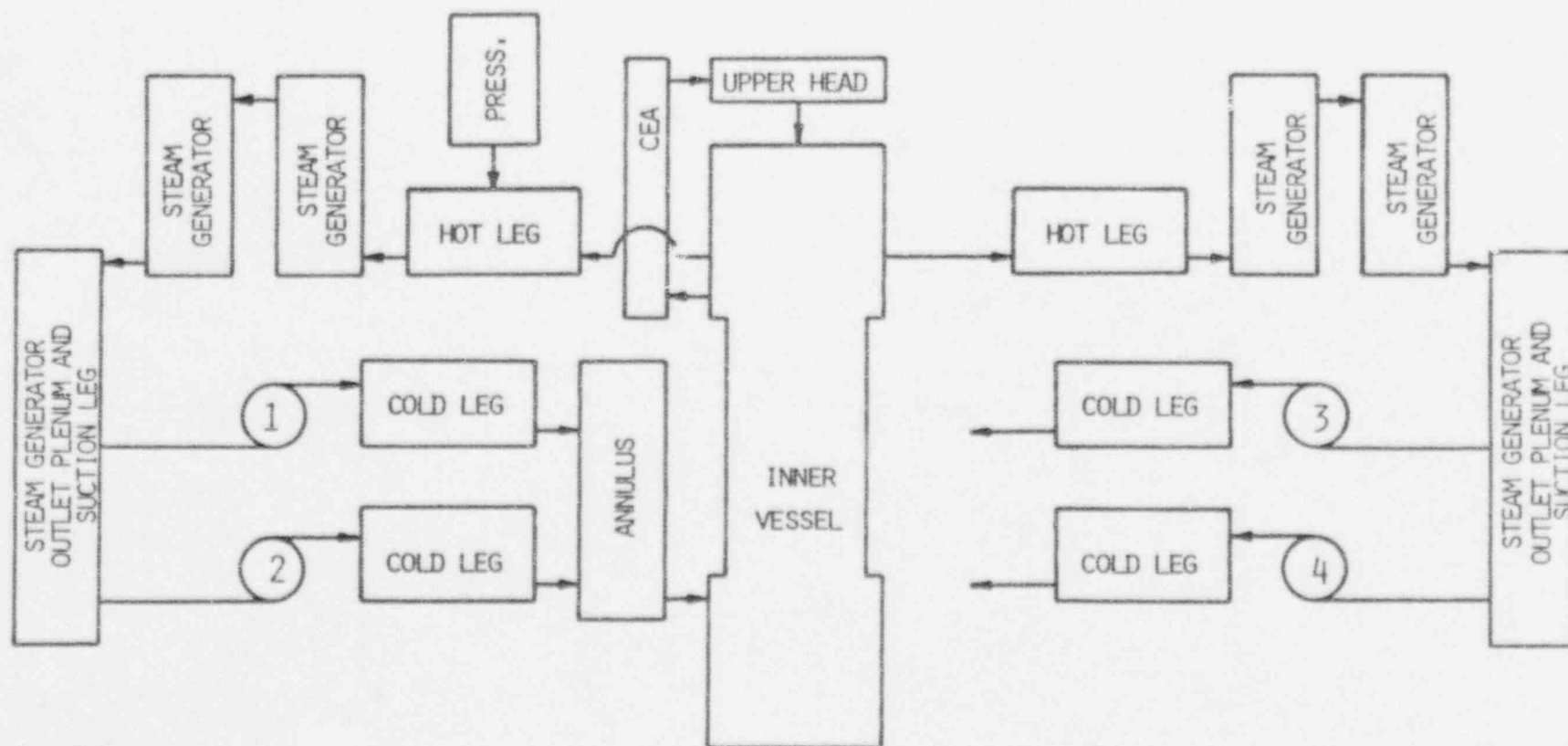


FIGURE 2
STEAM GENERATOR SECONDARY SYSTEM GEOMETRIC MODEL

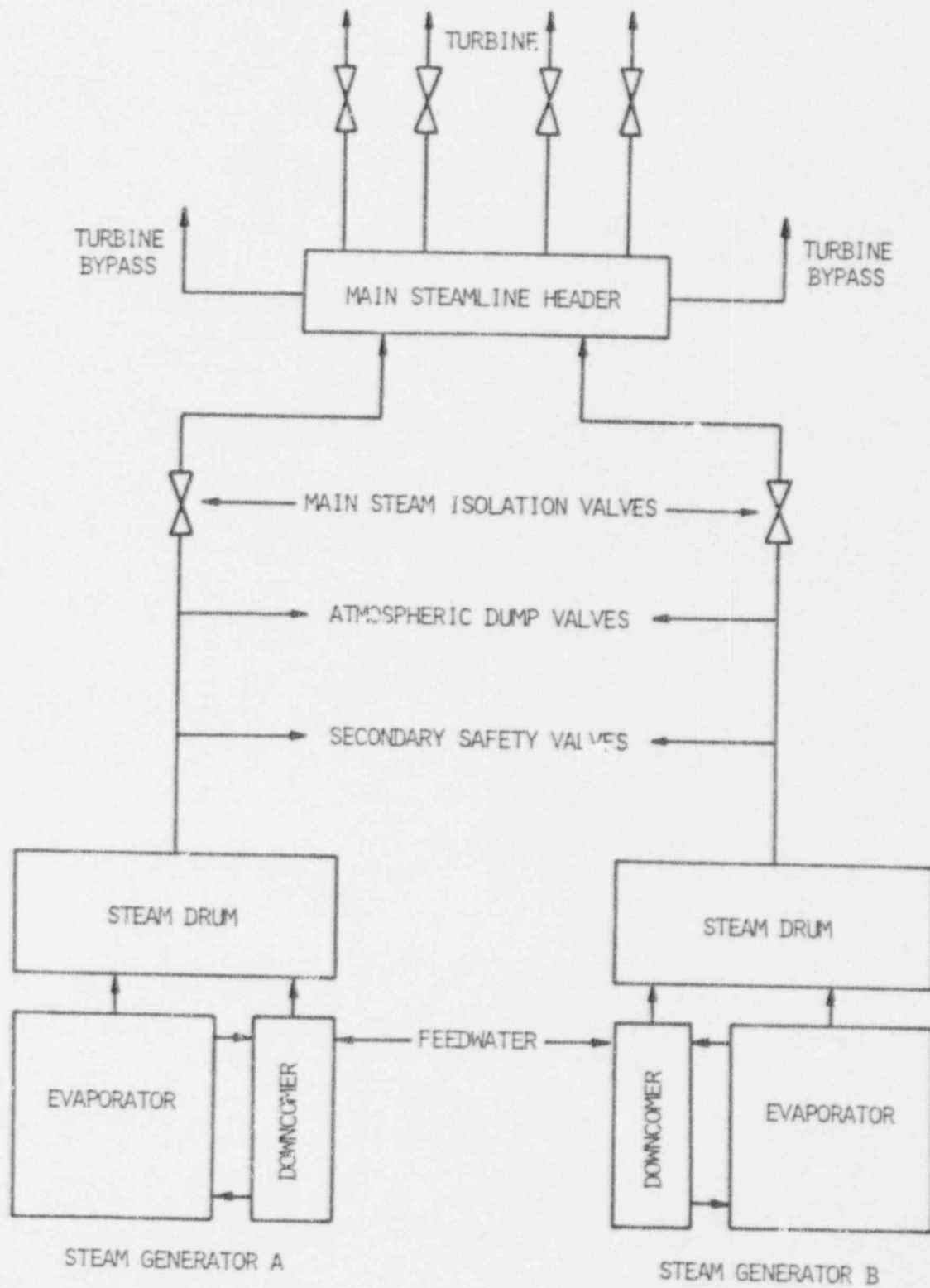


FIGURE 3
SMALL BREAK LOCA
(LOFT L3-1 TEST)

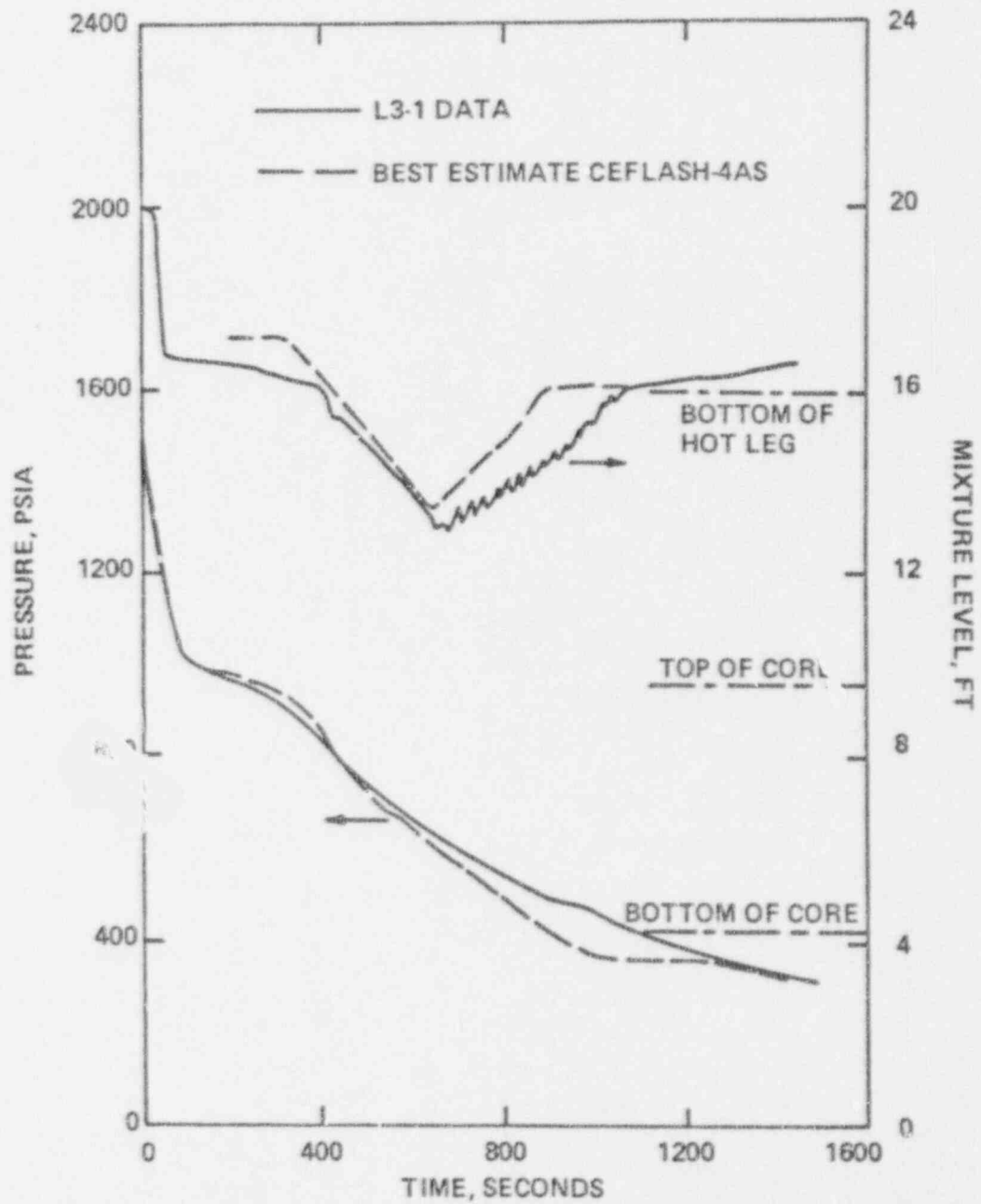


FIGURE 4
LOFT L3-6 PRIMARY SYSTEM PRESSURE

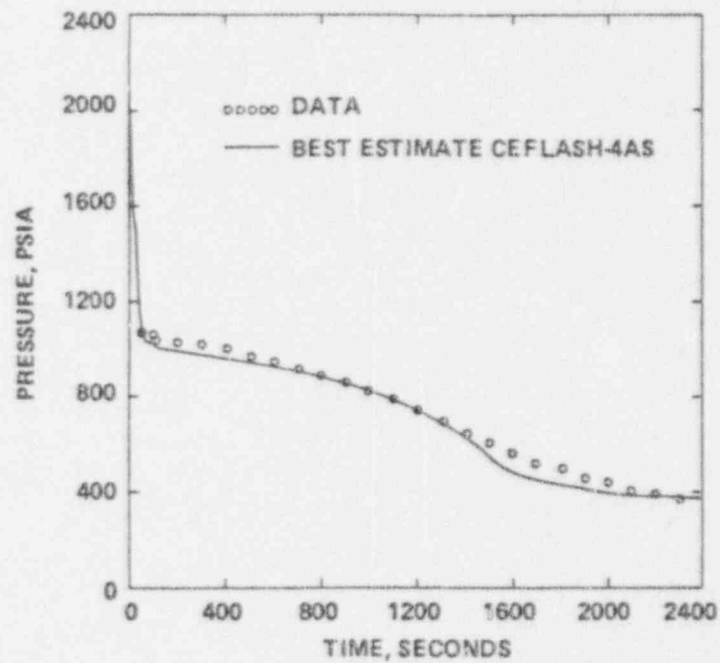


FIGURE 5
LOFT L3-6 PRIMARY SYSTEM MASS INVENTORY

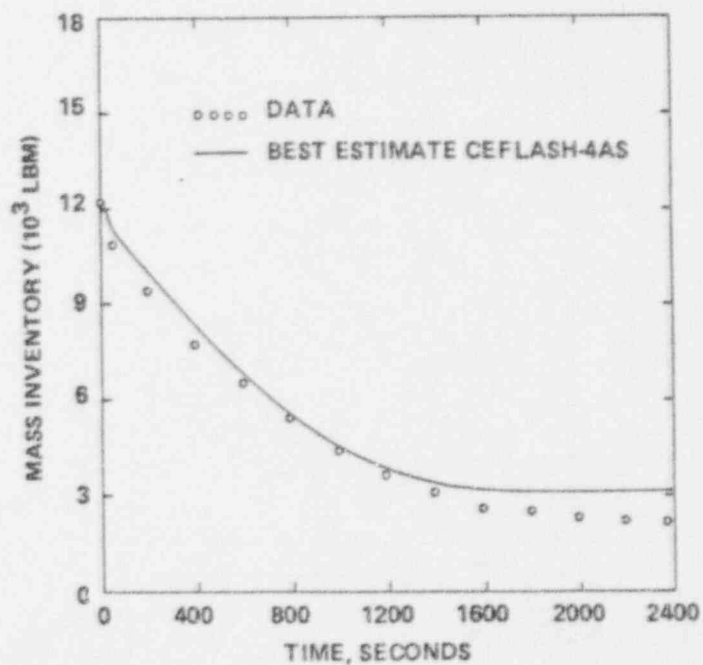


FIGURE 6
PRESSURIZER PRESSURE FOR PUMP TRANSIENT

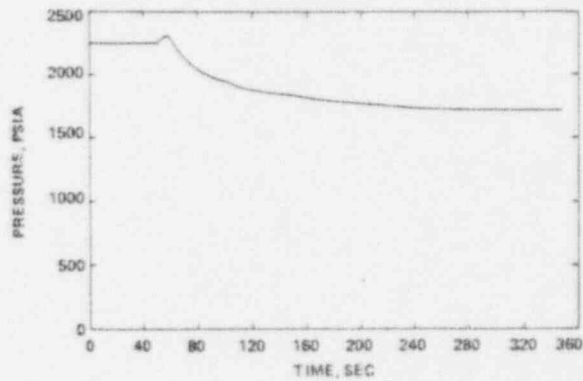


FIGURE 7
PRESSURIZER LEVEL FOR PUMP TRANSIENT

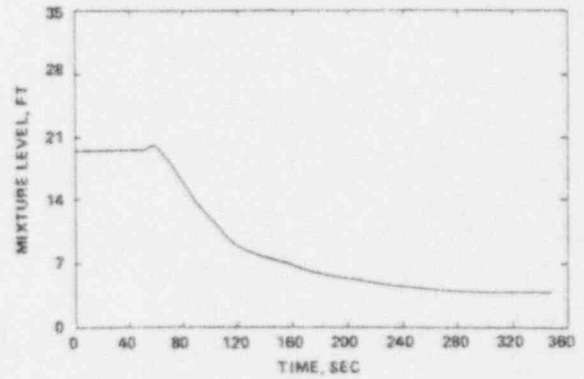


FIGURE 8
PUMP 1 FLOWRATE FOR PUMP TRANSIENT

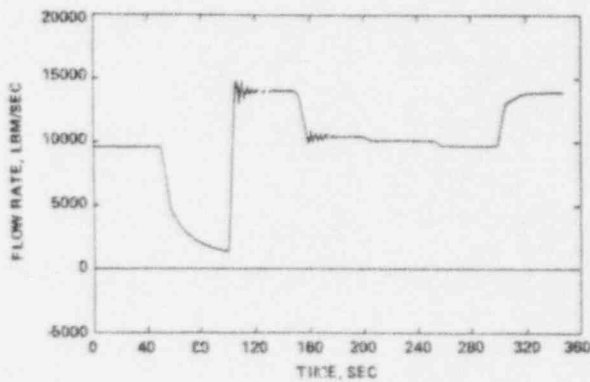


FIGURE 9
PUMP 2 FLOWRATE FOR PUMP TRANSIENT

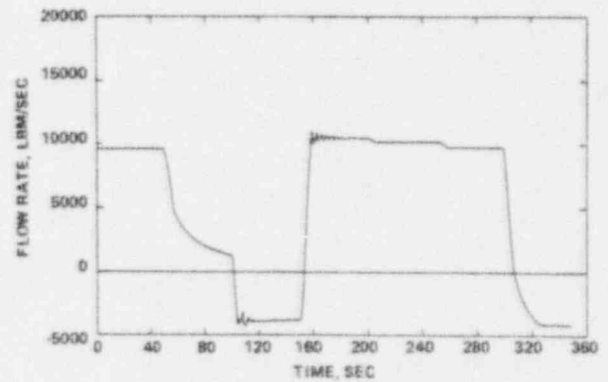


FIGURE 10
PUMP 3 FLOWRATE FOR PUMP TRANSIENT

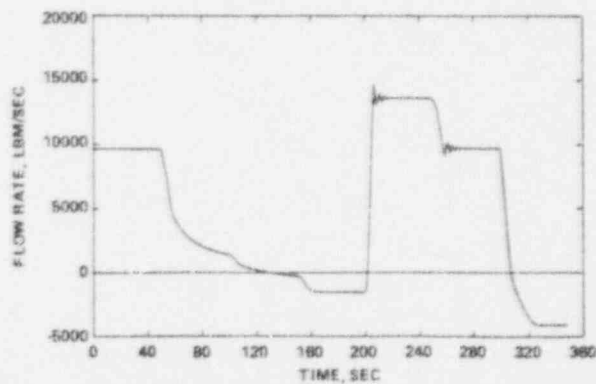


FIGURE 11
PUMP 4 FLOWRATE FOR PUMP TRANSIENT

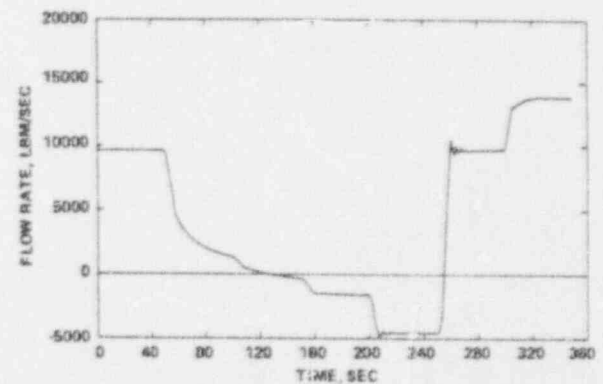


FIGURE 12
STEAM GENERATOR SECONDARY PRESSURE

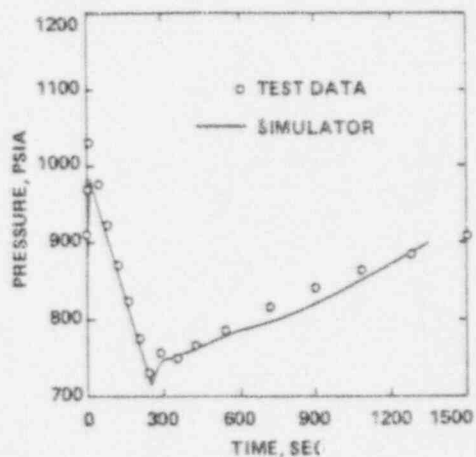


FIGURE 14
STEAM GENERATOR SECONDARY LEVEL

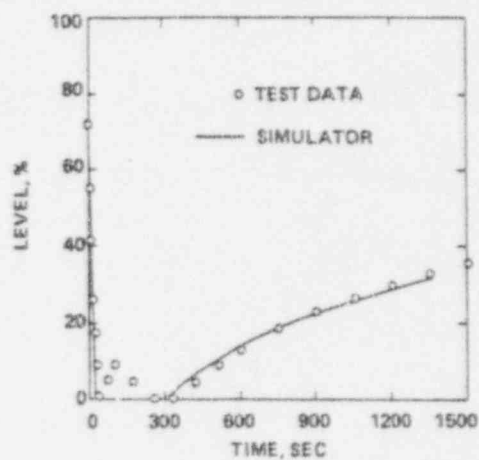


FIGURE 16
PRIMARY SYSTEM TEMPERATURES

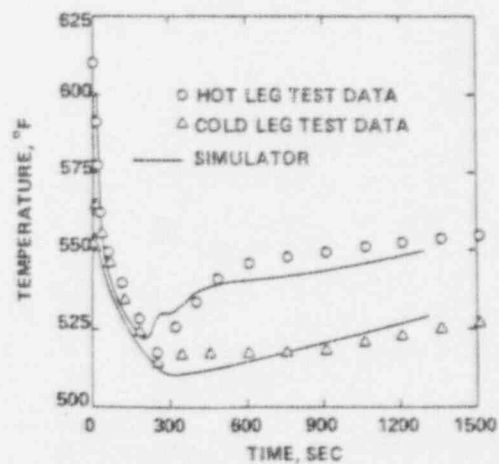


FIGURE 13
PRESSURIZER PRESSURE

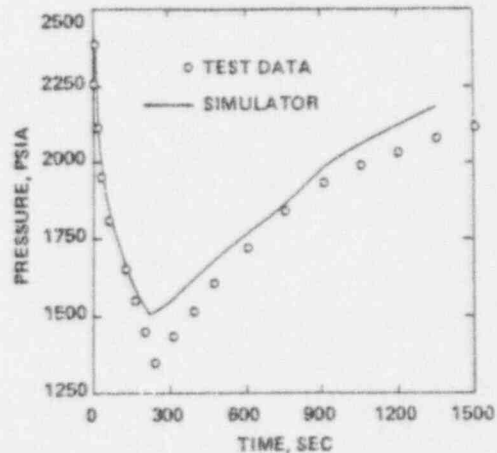


FIGURE 15
PRESSURIZER LEVEL

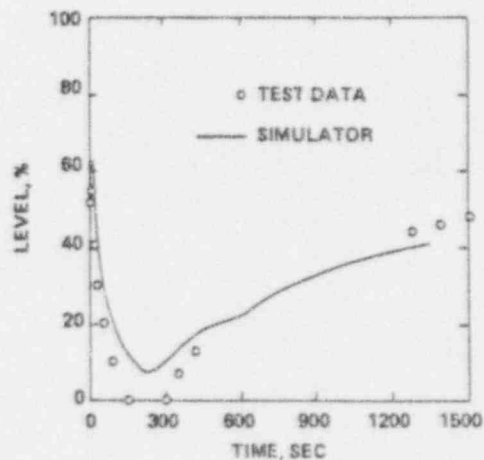


FIGURE 17
PRIMARY PRESSURE
FOR SMALL BREAK LOCA

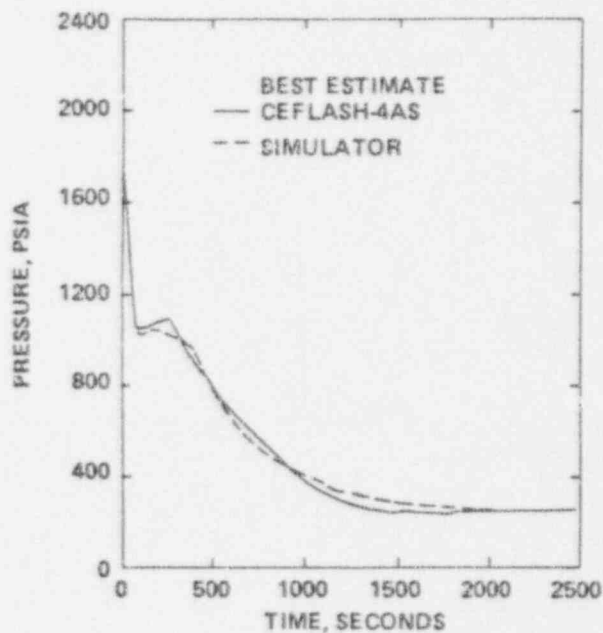


FIGURE 18
INNER VESSEL LEVEL
FOR SMALL BREAK LOCA

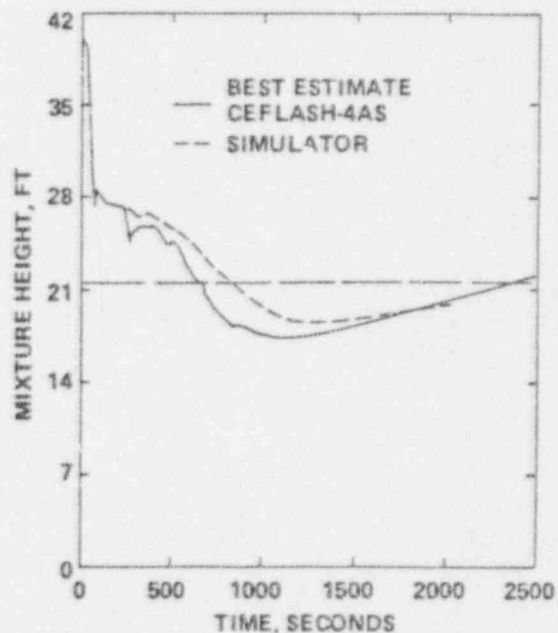
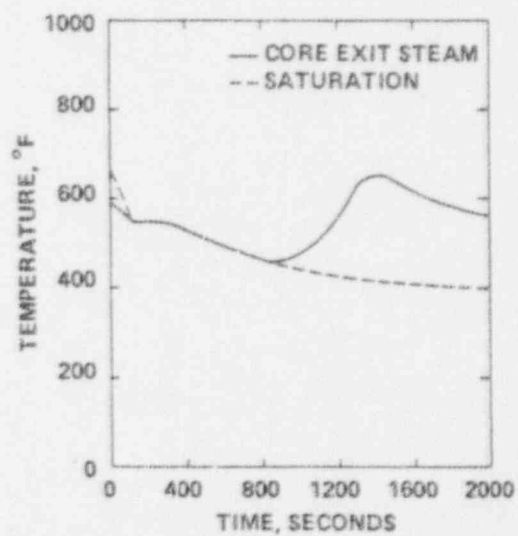


FIGURE 19
CORE EXIT STEAM TEMPERATURE
FOR SMALL BREAK LOCA



FAST AND SUPER-FAST NSSS SIMULATION USING IMPLICIT FIVE-EQUATION NONEQUILIBRIUM MODELS

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ABSTRACT

The thermal-hydraulic models and the integration technique of the Combustion Engineering Nuclear Transient Simulation (CENTS) code, an interactive and real-time simulation code, are described. The thermal-hydraulic model is a five-equation (two mass, two energy, and one momentum equation) nonequilibrium model. The integration of the conservation equations uses an implicit, convergent method which is a natural extension of and fully compatible with the equilibrium FLASH-4 integration. As such, applicability of the very efficient matrix inversion routine of the FLASH-4 code is demonstrated. Models for interactive, super-fast simulation of slow changing, long running transients are formulated. The application of the code to PWR problems is illustrated, and the fast-time capability is demonstrated with an example.

I. INTRODUCTION

The Combustion Engineering Nuclear Transient Simulation (CENTS) code is a real-time, interactive computer code. It simulates Nuclear Steam Supply System (NSSS) behavior for Pressurized Water Reactors (PWR) for a wide range of conditions, from normal operation to severe accidents with significant two-phase, including core uncover. Models are provided for the core (neutronics, heat transfer, and thermal-hydraulics), the primary system, the secondary system (steam generators, steam lines, and headers), and the control systems. In general, in terms of complexity and completeness, the CENTS models are comparable to those encountered in the industry's advanced plant simulation and design codes.

The CENTS code is also part of the Combustion Engineering (C-E) Setpoint Methodology, for Westinghouse designed plants, that calculates limiting safety system settings and limiting conditions for operation (on DNB and fuel centerline melting) (Reference 1). C-E's advanced simulator technology also uses the five-equation thermal-hydraulic model described here. Application of the model to full scope simulators is described in References 2 and 3. Upgrading existing simulators with this model allows them to satisfy the requirements of ANSI/ANS-3.5-1981 (Reference 4).

This paper describes the CENTS primary system thermal-hydraulic model, including the integration of the conservation equations. The primary system thermal-hydraulic model is a five-equation (two mass, two energy, and one momentum) thermal nonequilibrium model. The thermal-hydraulic formulation and the integration of the conservation equations are a natural extension of those used in the equilibrium, three-equation (one mass, one energy, and one momentum) FLASH-4 code (References 5 and 6).

The FLASH-4 code and its solution technique have been used in the past as the basis for the development of many three-equation thermal equilibrium codes like the RELAP4 family of codes and RETRAN. The significance of the FLASH-4 code is that it introduced a very stable integration technique and a very efficient solution of the discretized conservation equations. Of the two major nonequilibrium codes, TRAC and RELAP5 (References 7 and 8), only RELAP5 uses the integration scheme of the FLASH-4 code. However, its matrix solution differs from that of FLASH-4. RELAP5 is a five-equation model (two mass, one energy, and two momentum). The TRAC code is a six-equation model (two mass, two energy, and two momentum) whose formulation and solution technique are completely independent of the FLASH-4 model.

The CENTS thermal-hydraulic model is, on the other hand, fully compatible with the FLASH-4 integration technique (numerical integration and matrix solution). This compatibility allows application of the very efficient solution of the conservation equations of the FLASH-4 code to nonequilibrium problems.

The compatibility with the FLASH-4 solution is noted several times in this paper. In order to ease these references, the FLASH-4 solution technique is described briefly in Section II. Sections III - VI describe the CENTS thermal-hydraulic model, including the integration and solution of the conservation equations. Section VII describes modifications to the conservation equations to produce super-fast simulation that can be applied to simulation of slow changing, very long running transients. Section VIII illustrates the CENTS model capabilities by describing its application to PWRs, and demonstrates the fast-time capability with a specific example.

II. THE FLASH-4 SOLUTION TECHNIQUE (References 5 and 6)

The FLASH-4 code represents a thermal-hydraulic system by means of a hydraulic network consisting of nodes and flowpaths. The nodes enclose control volumes that represent the mixture mass and mixture energy in each node. The flowpaths are used to represent the mass flowrate between nodes (noncritical flowpaths), and external connections or leaks (critical flowpaths). Its thermal-hydraulic model is a three-equation thermal equilibrium model. It consists of conservation equations of mixture mass and mixture energy for each node, and mixture momentum for each noncritical flowpath. The mass flowrate for the critical flowpaths is calculated as a function of pressure and is used to represent choked flow for leak paths.

The above system of conservation equations can be written in vector form

$$\frac{dy}{dt} = f(t,y) \quad (II-1)$$

where

$$y = (W_1, W_2, \dots, M_1, M_2, \dots, U_1, U_2, \dots)^T$$

and

W = Flowpath flowrate

M = Node mixture mass

U = Node mixture momentum

t = time.

The above system of conservation equations is integrated using the implicit one-step method

$$[I - \Delta t f'(t, y)] \Delta y = \Delta t f(t, y) \quad (II-2)$$

where $f'(t, y)$ is the Jacobian of the function $f(t, y)$ and Δt is the time step. Equation (II-2) is obtained by linearization and backwards differentiation of Equation (II-1). Reference 1 demonstrates that this implicit one-step method is consistent, and thus convergent.

In order to solve the $K+2N$ by $K+2N$ linear system of discretized equations defined by Equation (II-2), where K is the number of noncritical flowpaths and N is the number of nodes, the FLASH-4 code explicitly calculates expressions for ΔM and ΔU as a function of ΔW . Substitution of these expressions into the corresponding flow equation yields a reduced $K \times K$ linear system of equations.

If, in addition, the flowpaths are numbered such that the disjoint chains¹ are numbered first, the reduced system of equations is of the form:

$$A \Delta W = Z$$

where A is a matrix of the form

$$A = \begin{bmatrix} D_1 & & & & P_1 \\ & D_2 & & & P_2 \\ & & & & \\ & & 0 & & D_s & P_s \\ Q_1 & Q_2 & & & Q_s & S \end{bmatrix} \quad (II-3)$$

The only nonzero elements in A occur in the blocks described above. Each of the D_i matrices is a tridiagonal matrix and is directly associated with each of the disjoint chains. This linear system of equations is solved using a block elimination, which for block type matrices like the one shown above, is significantly faster than Gaussian elimination.

¹A subnetwork is a chain if it is connected and every node of the chain is connected to at most two non critical flowpaths. See Reference 5 for a more detailed discussion of this one and related concepts.

III. THE CENTS THERMAL-HYDRAULIC MODEL

CENTS represents the thermal-hydraulic system by means of a hydraulic network consisting of nodes and flowpaths. The nodes again are control volumes that represent the mass and energy in the system. However, to accommodate the thermal nonequilibrium formulation, the mass and energy of each of the phases is represented separately in each node. The flowpaths are used to represent the mass flowrate between nodes (noncritical flowpaths) and the choked flow in external connections or leaks (critical flowpaths).

For each node i , let T_i denote the set of flowpaths for which i is the terminal node. Let I_i denote the set of flowpaths for which i is the initial node. Let $U_i = T_i \cup I_i$. Define the summation over the elements in U_i as follows.

$$\sum_{k \in U_i} = \sum_{k \in T_i} - \sum_{k \in I_i}$$

e.g.,

$$\sum_{k \in U_i} W_k = \sum_{k \in T_i} W_k - \sum_{k \in I_i} W_k$$

The mass and energy conservation equations for each node i are the following:

i. Conservation of liquid mass

$$\frac{dM_{l,i}}{dt} = \sum_{k \in U_i} (1-x_k) W_k + W_{\text{cond},i} - W_{\text{evap},i} \quad (\text{III-1})$$

ii. Conservation of mixture mass

$$\frac{dM_i}{dt} = \sum_{k \in U_i} W_k \quad (\text{III-2})$$

iii. Conservation of mixture energy

$$\frac{dU_i}{dt} = \sum_{k \in U_i} W_k h_k + Q_i \quad (\text{III-3})$$

iv. Conservation of steam total enthalpy

$$\begin{aligned} \frac{dH_{st,i}}{dt} = & \sum_{k \in U_i} x_k W_k h_{st,k} + Q_{st,i} - W_{\text{cond},i} h_f \\ & + W_{\text{evap},i} h_g \end{aligned} \quad (\text{III-4})$$

The nomenclature for these equations is

M_l, M	= Liquid, total mass
U	= Total internal energy
H	= Total steam enthalpy
Q, Q_{st}	= Total, steam heat rate
W	= Mass flowrate
h, h_{st}	= Flowpath mixture, steam enthalpy
x	= Flowpath quality
h_f, h_g	= Saturation liquid, steam enthalpies
W_{cond}	= Condensation rate
W_{evap}	= Evaporation rate.

The momentum equation for flowpath k connecting nodes i and j is of the form:

$$L_k \frac{dW_k}{dt} = P_i - P_j - F W_k |W_k| + E + G \quad (III-5)$$

where

P_i, P_j	= Upstream, downstream pressure
F	= Frictional coefficient (the code calculates separately geometric losses and Reynolds dependent frictional losses. They are lumped together for the present discussion).
E	= Elevation head
G	= Pump head (for pump paths)
L	= Flowpath inertia.

The critical flow equation for leaks is of the form:

$$W_k = W(P, h) \quad (III-6)$$

where $W(P, h)$ is a functional expression for choked flow.

Assuming that the network consists of N nodes and K noncritical flowpaths, the above system of conservation equations can be written in the form:

$$\frac{dy}{dt} = f(t, y) \quad (III-7)$$

where

$$y = (W_1, \dots, W_K, M_{l,1}, \dots, M_{l,N}, M_1, \dots, M_N, U_1, \dots, U_N, H_{st1}, \dots, H_{st,N})^T$$

This system is coupled, since the pressures in the momentum equation are functions of the nodal liquid mass, mixture mass, mixture energy, and steam total enthalpy, i.e.,

$$P = P(M_l, M, U, H_{st})$$

The functional dependency that allows derivation of the expression above is given later in Section V. In addition, the critical mass flowrates for the leak paths in Equations (III-1) - (III-4) are a function of the pressure and node enthalpy (Equation III-6), further increasing coupling of the system.

IV. INTEGRATION OF THE CENTS CONSERVATION EQUATIONS

Equation (III-7) is integrated using the FLASH-4 implicit one-step method

$$[I - \Delta t f'(t, y)] \Delta y = \Delta t f(t, y) \quad (IV-1)$$

described in Section II. As stated in Section II, this implicit one-step method is consistent, and thus convergent.

Equation (IV-1) is a linear system of $K+4N$ equations in $K+4N$ unknowns. The solution of this system of equations can be obtained rather efficiently by first eliminating the terms ΔM_ℓ , ΔM , ΔU , and ΔH_{st} from the discretized flow equations (a comparable reduction to the one that is done in the FLASH-4 code for an equilibrium formulation). This elimination reduces the system to a linear system of K equations in K unknowns, $\Delta W_1, \dots, \Delta W_K$.

In order to illustrate this reduction, let:

$$F^{(i)} = (f_{K+1}(t, y), f_{K+N+1}(t, y), f_{K+2N+1}(t, y), f_{K+3N+1}(t, y))^T$$

denote the vector consisting of the right hand sides of Equations (III-1), (III-2), (III-3), and (III-4) for node i . Let the set U_i which describes the set of flowpaths connected to node i , be expressed as $U_i = U_{i,1} \cup U_{i,2}$, where

$U_{i,1}$ is the set of noncritical paths which are connected to node i , and $U_{i,2}$ is the set of critical paths connected to node i . Then the linearized equations for $\Delta M_{\ell,i}$, ΔM_i , ΔU_i and $\Delta H_{st,i}$ can be written in matrix form as follows.

$$[I - \Delta t C^{(i)}] (\Delta S^{(i)}) + R^{(i)} \Delta W = \Delta t F^{(i)} \quad (IV-2)$$

where $\Delta S^{(i)} = (\Delta M_{\ell,i}, \Delta M_i, \Delta U_i, \Delta H_{st,i})^T$,

$C^{(i)}$ is a 4×4 matrix with entries

$$C_{m,1}^{(i)} = k_{EU,2,i} \alpha_k V_m^{(k)}, \quad C_{m,2}^{(i)} = k_{EU,2,i} \beta_k V_m^{(k)}$$

$$C_{m,3}^{(i)} = k_{EU,2,i} \gamma_k V_m^{(k)}, \quad C_{m,4}^{(i)} = k_{EU,2,i} \delta_k V_m^{(k)}$$

for $m = 1, \dots, 4$, and

$$\alpha_k = \frac{\partial W_k}{\partial P} \frac{\partial P}{\partial M_\ell} + \frac{\partial W_k}{\partial h} \frac{\partial h}{\partial M_\ell}, \quad \beta_k = \frac{\partial W_k}{\partial P} \frac{\partial P}{\partial M} + \frac{\partial W_k}{\partial h} \frac{\partial h}{\partial M},$$

$$\gamma_k = \frac{\partial W_k}{\partial P} \frac{\partial P}{\partial U} + \frac{\partial W_k}{\partial h} \frac{\partial h}{\partial U}, \quad \delta_k = \frac{\partial W_k}{\partial P} \frac{\partial P}{\partial H_{st}} + \frac{\partial W_k}{\partial h} \frac{\partial h}{\partial H_{st}},$$

$$\begin{aligned} V^{(k)} &= (V_1^{(k)}, V_2^{(k)}, V_3^{(k)}, V_4^{(k)})^T \\ &= (1 - x_k, 1, h_k, x_k h_{st,k})^T \end{aligned}$$

The matrix $R^{(i)}$ is a $4 \times K$ matrix with entries

$$R_{m,k}^{(i)} = \begin{cases} V_m^{(k)}, & \text{if } k \in T_i, \\ -V_m^{(k)}, & \text{if } k \in I_i, \\ 0, & \text{otherwise.} \end{cases}$$

Let

$$B^{(i)} = [I - \Delta t C^{(i)}]^{-1}. \quad (IV-3)$$

Then, it follows from Equation (IV-1) that

$$\Delta M_{\ell,i} = \eta_1^{(i)} + \Delta t \sum_{k \in U_1} [(1-x_k) B_{11}^{(i)} + B_{12}^{(i)} + h_k B_{13}^{(i)} + x_k h_{st,k} B_{14}^{(i)}] \Delta W_k \quad (IV-4)$$

$$\Delta M_i = \eta_2^{(i)} + \Delta t \sum_{k \in U_1} [(1-x_k) B_{21}^{(i)} + B_{22}^{(i)} + h_k B_{23}^{(i)} + x_k h_{st,k} B_{24}^{(i)}] \Delta W_k \quad (IV-5)$$

$$\Delta U_i = \eta_3^{(i)} + \Delta t \sum_{k \in U_1} [(1-x_k) B_{31}^{(i)} + B_{32}^{(i)} + h_k B_{33}^{(i)} + x_k h_{st,k} B_{34}^{(i)}] \Delta W_k \quad (IV-6)$$

$$\Delta H_{st,i} = \eta_4^{(i)} + \Delta t \sum_{k \in U_1} [(1-x_k) B_{41}^{(i)} + B_{42}^{(i)} + h_k B_{43}^{(i)} + x_k h_{st,k} B_{44}^{(i)}] \Delta W_k \quad (IV-7)$$

where the vector $\eta^{(i)} = (\eta_1^{(i)}, \eta_2^{(i)}, \eta_3^{(i)}, \eta_4^{(i)})^T$ equals

$$\eta^{(i)} = \Delta t B^{(i)} F^{(i)}.$$

Now assume that the k th flowpath joins the i th and j th node. The corresponding discretized momentum equation is:

$$\begin{aligned} & (1 + \Delta t \frac{2F_k |W_k|}{L_k}) \Delta W_k \\ & - \frac{\Delta t}{L_k} (\sigma_i \Delta M_{\ell,i} + \rho_i \Delta M_i + \epsilon_i \Delta U_i + \mu_i \Delta H_{st,i} \\ & \quad - \sigma_j \Delta M_{\ell,j} - \rho_j \Delta M_j - \epsilon_j \Delta U_j - \mu_j \Delta H_{st,j}) \\ & = \Delta t f_k(y) \end{aligned} \quad (IV-8)$$

where

$$\sigma = \frac{\partial P}{\partial M_k}, \quad \rho = \frac{\partial P}{\partial M}, \quad \xi = \frac{\partial P}{\partial U}, \quad \mu = \frac{\partial P}{\partial H_{st}}.$$

Substitution of Equations (IV-4), (IV-5), (IV-6), and (IV-7) into Equation (IV-8) yields a reduced $K \times K$ system of linear equations of the form:

$$A \Delta W = Z \quad (IV-9)$$

where

$$Z_k = \Delta t \left[f_k(y) + \frac{1}{L_k} (\sigma_i n_1^{(i)} + \rho_i n_2^{(i)} + \xi_i n_3^{(i)} + \mu_i n_4^{(i)} - \sigma_j n_1^{(j)} - \rho_j n_2^{(j)} - \xi_j n_3^{(j)} - \mu_j n_4^{(j)}) \right]$$

$$A_{k,k} = 1 + \Delta t \frac{2F_k |W_k|}{L_k} - A_{k,k}^*$$

$$A_{k,n} = A_{k,n}^* \text{ if } k \neq n.$$

The entries $A_{k,n}^*$ are defined as follows (in matrix form):

$$i. \text{ If } n \in T_i \text{ and } n \in I_j, A_{k,n}^* = \frac{(\Delta t)^2}{L_k} [[X^{(i)}]^T B^{(i)}_Y - [X^{(j)}]^T B^{(j)}_Y]$$

$$ii. \text{ If } n \in I_i \text{ and } n \in T_j, A_{k,n}^* = - \frac{(\Delta t)^2}{L_k} [[X^{(i)}]^T B^{(i)}_Y - [X^{(j)}]^T B^{(j)}_Y]$$

$$iii. \text{ If } n \in T_i \text{ and } n \notin I_j, A_{k,n}^* = \frac{(\Delta t)^2}{L_k} [X^{(i)}]^T B^{(i)}_Y$$

$$iv. \text{ If } n \in I_i \text{ and } n \notin T_j, A_{k,n}^* = - \frac{(\Delta t)^2}{L_k} [X^{(j)}]^T B^{(j)}_Y$$

$$v. \text{ If } n \in I_j \text{ and } n \notin T_i, A_{k,n}^* = \frac{(\Delta t)^2}{L_k} [X^{(i)}]^T B^{(i)}_Y$$

$$vi. \text{ If } n \in T_j \text{ and } n \notin I_i, A_{k,n}^* = \frac{(\Delta t)^2}{L_k} [X^{(j)}]^T B^{(j)}_Y$$

where

$$x^{(i)} = (\sigma_i, \rho_i, \epsilon_i, \nu_i)^T$$

$$Y = (1 - x_k, 1, h_k, x_k h_{st,k})^T \text{ and}$$

$$B^{(i)} = [I - \Delta t C^{(i)}]^{-1} \text{ (Equation IV-3).}$$

The $K \times K$ coefficient matrix A for the reduced system (IV-9) has possibly nonzero elements in exactly the same locations as the corresponding FLASH-4 coefficient matrix for the equilibrium reduced system. Thus, if the flowpaths are numbered such that the disjoint chains are numbered first, as described in Section II, the coefficient matrix A is of the form described in Equation (II-3). Therefore the FLASH-4 block elimination solution of Equation (IV-9) can be used with no changes.

V. CALCULATION OF THE NODE AVERAGE CONDITIONS

Given the node mixture and liquid mass (M and M_l), the mixture energy (U), the steam total enthalpy (H_{st}), and the node volume (V), then the node pressure (P), the steam mass (M_{st}), the liquid and steam enthalpies (h_l and h_{st}), and the liquid and steam specific volumes (v_l and v_{st}) satisfy the following system of equations.

$$M_{st} = M - M_l \quad (V-1)$$

$$h_{st} = H_{st}/M_{st} \quad (V-2)$$

$$h = U/M + VP \quad (V-3)$$

$$V = M_l/v_l + M_{st}/v_{st} \quad (V-4)$$

$$v_l = v(P, h_l) \quad (V-5)$$

$$v_{st} = v(P, h_{st}) \quad (V-6)$$

where the expression $v(P, h)$ is the specific volume equation of state. Substitution of Equations (V-1), (V-2), (V-5) and (V-6) into Equations (V-3) and (V-4) yields a nonlinear system of equations in P and h_l which is then solved using Newton's method.

VI. THE CENTS CONSTITUTIVE PACKAGE

The emphasis during the development of CENTS was on both speed and accuracy. In order not to compromise the code speed, a complete but relatively simple constitutive package was used. The constitutive equations include all major modes of heat transfer and mass transfer between phases i.e., condensation on the liquid surface, condensation on walls, condensation of bubbles, condensation on the injected fluid, evaporation by wall heat, by flashing, etc. The mass-heat transfer is calculated using correlations from the literature (TRAC, RELAP5). No attempt was made at implementing flow regimes. However, since in terms of its attention to detail in the formulation of the heat and mass transfer between phases the CENTS thermal-hydraulic formulation is comparable to that of TRAC, a constitutive package comparable in sophistication to that one could be implemented into CENTS.

VII. SUPER FAST SIMULATION - THE FAST-TIME MODEL

In interactive real-time simulation it is important to have the capability to simulate long running transients (e.g., a complete heatup or cooldown) much faster than real-time. This capability allows the user to observe the system response in a reasonable time span. This capability also allows the user to easily and quickly generate initial conditions from a vastly different initial state, for example, generation of initial conditions for shutdown cooling starting from a 100% power system state.

A fast-time model with the capability to run much faster than the base CENTS model and without losing any of the geometric detail is available in the CENTS code. The model is not claimed to be the exact replica of the base model running M times faster. Rather, it is a simulation which approximates the accelerated response of the system, without using large time steps that would destabilize the solution.

The fast-time model employs the CENTS base model (the five-equation thermal-hydraulic model) with the same base time step, and with a diffusion model that incorporates into the system the integrated effects of various parameters during the large time steps. For example, the integrated effect of heat addition to the loops during a large time step can be expressed as follows

$$\int_{\Delta t_{\text{large}}} \dot{Q} dt = \int_{\Delta t_{\text{small}}} \dot{Q} dt + \int_{\Delta t_{\text{large}} - \Delta t_{\text{small}}} \dot{Q} dt. \quad (\text{VII-1})$$

Integration of the conservation equations with the small time steps only incorporates into the system the net heat addition described by the first term on the right hand side of Equation (VII-1). In order to simulate the net heat addition described by the second term on the right hand side of Equation (VII-1), a diffusion term $d_i(Q)$ is added to the conservation of energy equation for each node. The diffusion terms are defined such that

$$\int_{\Delta t_{\text{small}}} d_i(Q) = \int_{\Delta t_{\text{large}} - \Delta t_{\text{small}}} \dot{Q} dt. \quad (\text{VII-2})$$

Similarly, diffusion terms for fluid sources/sinks, or other effects are defined and included in each of the conservation equations.

For semi-isolated components, like the pressurizer, very little, if any, diffusion of the various parameters occurs to the main system (i.e., the loops). Thus, the diffusion terms for parameters defined for semi-isolated components, e.g., pressurizer heater heat addition, still satisfy Equation (VII-2), but are defined as non-zero only in the same component.

It is not always possible to reproduce the exact form of the diffusion terms in a precise manner. However, experience has shown that "reasonable" expressions, e.g., weighted terms based on total mass, fluid flow, etc., produce adequate approximations and reasonable response.

VIII. EXAMPLES

A typical C-E designed plant is represented with the CENTS code using the nodalization shown in Figure 1. Similar representations are used for plants of Westinghouse design. With this nodalization and time steps of the order of one second, CENTS is capable of simulating NSSS transients rapidly, accurately and with remarkable stability. Typical transients execute between three and four times faster than real-time on the Perkin-Elmer 3244 minicomputer.

Transient comparisons to plant data and design code calculations have been documented elsewhere (References 2 and 3), and are not repeated here.

The fast-time capability is illustrated here by means of a plant heatup. The system starts cold (hot leg temperature 372°K (210°F), pressurizer pressure 689.5 kPa (100 psia) with the shutdown cooling system isolated, all four reactor coolant pumps running, and the pressurizer heaters turned on. With the pressurizer level control in automatic mode, the plant heatup was run once using the regular one second time steps (without fast time) and a second time using the fast-time feature scaled to run sixty times faster. Figure 11, shows a comparison of the pressurizer pressure, pressurizer temperature, and hot leg temperature for these two cases during the first 10,000 seconds of the transient. The relative closeness of the two calculations demonstrates the power of the fast time model. Using the time scaling ratio of 60, the complete heatup, to approximately 20,000 seconds, was simulated live, interactively, in approximately 2 minutes of computer time on the Perkin-Elmer 3244.

IX. CONCLUSIONS

The FLASH-4 integration technique for a three-equation equilibrium model has been successfully extended to a five-equation model for nonequilibrium conditions. The same explicit solution technique used to reduce the discretized linear system of equations for an equilibrium system to a block type matrix system applies for the five thermal nonequilibrium equations system and produces a matrix of the same form. Consequently the FLASH-4 block elimination solution technique is also applicable for the thermal nonequilibrium formulation. The nonequilibrium formulation integration technique shares the convergence characteristics of the equilibrium formulation.

For slowly changing transients, introduction of diffusion terms in the conservation equations extends the five-equation model to a fast time mode. The resulting fast-time model allows up to a 60 fold increase in computation speed and gives very satisfactory agreement with the results of calculations done with the base model.

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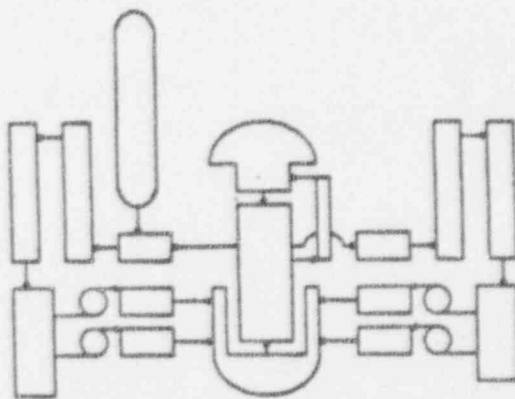


FIGURE I. Typical C-E Plant CENTS Flow Network

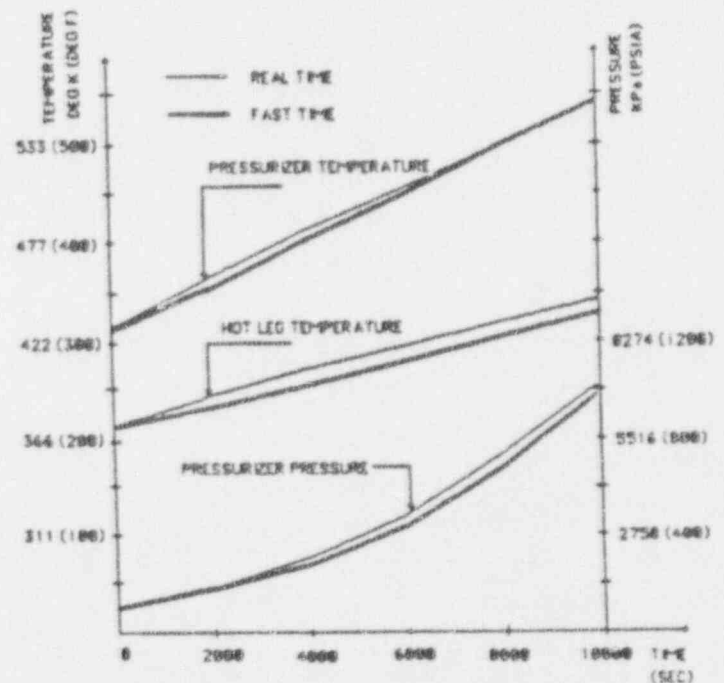


FIGURE II. Heatup Parameters: Real Time vs Fast Time

NATURAL CIRCULATION COOLDOWN USING THE COMBUSTION ENGINEERING SIMULATOR

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ABSTRACT

A full scope simulator with thermal-hydraulic models based on first principles models is shown to reproduce observed plant behavior for a natural circulation cooldown with upper head voiding. Inclusion of nonequilibrium, phase-separation models allows reproduction of the two-phase fluid behavior when a bubble is drawn in the upper head of the reactor vessel. Level oscillations in the upper head and pressurizer due to operator actions including refill of the upper head are reproduced. The ability to use the simulator instead of the plant for operator training and to extend the training beyond that possible with the plant alone is shown with examples.

I. INTRODUCTION

The Nuclear Regulatory Commission (NRC) recommendations for Residual Heat Removal (RHR) systems have produced a substantial need for operator training to perform natural circulation cooldowns. The NRC now recommends that a nuclear power plant be capable of being cooled to cold shutdown conditions using only safety-grade equipment concurrent with a loss of offsite power and an assumed single failure.¹ Due to loss of offsite power, the cooldown will be by natural circulation. The NRC further suggests that new operating licensees conduct a low-power test to provide "hands-on" training.² As a result, nuclear power plants are performing natural circulation tests at low power conditions prior to escalation to full power.

The natural circulation tests train operators to recognize and evaluate plant behavior during natural circulation and to operate the plant appropriately during natural circulation conditions. Operators are trained to control the cooldown rate and pressurizer pressure so as to maintain a subcooled margin during cooldown. This is done by manipulation of charging flow, secondary steaming rate, and other plant systems. Tests are performed to initiate, verify and maintain natural

circulation as primary pressure decreases. Variations of plant conditions include lack of the pressurizer heaters and asymmetric cooldowns (one steam generator isolated).

Alternatives to performing operator training with the power plant are desirable. Reactor test programs are time consuming and expensive to perform. Subsequent operator training is difficult to do with a power plant once it is in commercial operation. Further, the range of feasible training situations is limited because of the risk of damage to plant equipment, e.g., cooldown with loss of both offsite and onsite alternating current.

Use of a full scope reactor training simulator for training in performing natural circulation cooldowns is an alternative to training with the power plant. Satisfactory training on the simulator requires that it have a sufficient level of fidelity to the plant behavior. This, in turn, requires that the simulator use first principles models which reproduce the dominant physical processes which occur during the transients of interest.

This paper demonstrates the simulation and training capabilities of a full scope simulator developed by Combustion Engineering on the basis of first principles models. The thermal-hydraulic models used in the C-E simulator are described briefly. Their capability to simulate the natural circulation cooldown behavior of a Pressurized Water Reactor (PWR) is demonstrated by comparison of the plant behavior predicted by the simulator to plant data for a natural circulation cooldown. The transient presented includes formation of a void in the upper head of the reactor vessel and a procedure for elimination of the upper head void. Finally, use of the models for operator training is discussed and demonstrated with three procedures for cooldown of the upper head.

II. SIMULATOR THERMAL-HYDRAULIC MODEL

The simulator is designed to provide

realistic thermal-hydraulic responses and instrument indications for a full range of operating conditions. The models are based on those used for PWR design codes -- CEFLASH-4AS for the primary system or Reactor Coolant System (RCS) and the Long Term Cooling (LTC) code for the steam generator secondary system.^{5,4}

The primary system thermal-hydraulic response is modeled by a node and flowpath network. The nodes enclose control volumes which represent the mass and energy. Nodes are provided for the major components in the C-E PWR reactor primary system -- inner vessel, upper head, control element assembly (CEA), guidetubes, two hot legs, a pressurizer, two steam generators (separate hot and cold sides of the U-tubes), two combined outlet plenum and pump suction legs, four cold legs and reactor coolant pumps, and the annulus or downcomer of the reactor vessel. Flowpaths connecting the nodes represent the fluid momentum and have no volume. Additional non-momentum flowpaths are provided for addition or removal of coolant by various reactor systems.

The RCS thermal-hydraulic model is formulated with five one-dimensional conservation equations. The conservation variables are mixture (liquid and steam) mass, liquid mass, mixture energy, steam energy and mixture momentum. The model incorporates slip.⁵ The conservation equations are integrated implicitly by means of simultaneous solution of the linearized, discretized conservation equations. The mass and energy in the liquid and steam regions of each node are used with water property correlations and the nodal volume to determine the resulting state variables (pressure, phase enthalpies and phase temperatures).

The simulator models phase separation within a node into a separate steam region and a liquid or two-phase region. Phase separation is calculated from experimentally based drift flux correlations.^{6,7} Phase-separated nodes have a discrete two-phase level which is used to determine fluid quality in flowpaths. The simulator model automatically calculates the thermodynamic fluid state for each node. This includes equilibrium conditions for homogeneous nodes and nonequilibrium for phase-separated nodes. The nonequilibrium states include subcooled liquid-saturated steam, saturated liquid - superheated steam, and subcooled liquid - superheated steam.

A core heat transfer model includes forced convection and quiescent pool boiling with

appropriate boiling curves. The mode of heat transfer is determined dynamically for each axial fuel rod node. A radial temperature distribution in the fuel and cladding is provided for each axial node. Reactor power is calculated by a three-dimensional reactor core neutronic model as needed.

The steam generator secondary system model includes nonequilibrium and recirculation effects. Each steam generator is represented by a downcomer, an evaporator and a steam drum with a common steam line header for both generators. Discrete fluid levels are produced in the evaporator and downcomer. Forward and reverse, level dependent heat transfer with appropriate boiling curves are represented. Mass and energy balances are provided for each node. The flow between the downcomer and the evaporator and the main steam flows are calculated using momentum balances.

The thermal-hydraulic models run faster than real time on a Perkin-Elmer 3244 minicomputer which allows real time simulation of the transients. Further details of the thermal-hydraulic and neutronic models are given elsewhere.^{8,9} Examples of their validation for several transients are also given in these references.

III. SIMULATION OF PLANT COOLDOWN TRANSIENT

The performance of the simulator thermal-hydraulic models for a natural circulation transient is demonstrated by a comparison of plant behavior calculated by the simulator to measured plant data. The transient includes natural circulation cooldown of the plant, formation of a bubble in the upper vessel head, and subsequent cooldown with cyclic drain and fill to cool the upper head. It tests the response of the simulator to a wide range of system conditions, thermal-hydraulic states, and operator actions. Successful simulation of upper head drain and fill requires sophisticated two-phase models including phase-separation, non-equilibrium conditions, steam condensation, flow into and out of the upper head and wall heat transfer. The C-E simulation presented here includes all of these processes.

On June 11, 1980 an electrical failure caused a component cooling water isolation valve to shut off cooling water flow to the seals of all four reactor cooling pumps (RCP) at a plant with a C-E Nuclear Steam Supply System. The plant was operating at nearly full power with letdown and charging in service. Loss of RCP cooling required the operators to

shut down the reactor at 2:33 am. The reactor was then cooled down by natural circulation cooling of the core. A detailed description of the transient is given in Reference 10.

Simulation for seven hours of the transient is provided. The first four hours cover plant trip and the initial cooldown. The last three hours cover formation of the upper head bubble and several pressurizer level oscillations. The simulation is performed interactively with the simulator software. Due to a lack of data on operation of the plant systems, their operation for the simulation was guided by the apparent operation actions. Timing of the changes was dictated by the point at which conditions in the simulation require the action. A summary of the principle events used in the simulation is given in Table 1. It closely follows the recorded events in the plant transient.

Simulation of the transient began at 2:30 am to establish initial conditions. Following plant trip at 2:33 am, reactor pressure declined rapidly from 15.5 mP (2250 psia), Figure 1. Pressurizer level dropped rapidly from midrange, Figure 2. Reactor coolant temperature in the hot leg, T_{hot} , dropped rapidly, Figure 3, until the reactor coolant pumps (RCP) were tripped at 2:35 am. Reactor pressure, pressurizer level, and hot leg temperature rose briefly until one pump was restarted at 2:38 am, then they resumed their previous decline. When the pump was tripped at 2:39 am, pressure, pressurizer level and hot leg temperature rose due to heat up of fluid in the loop. By 2:43 am, natural circulation was established which ended the rise in pressure, pressurizer level and hot leg temperature. The temperature difference between the hot and cold legs was about 14°K (25°F). Through the first 30 minutes of the transient, the simulation reproduces the major features of the transient. Small variations in hot and cold leg temperatures, pressure, and pressurizer level are not reproduced since the details of the causative operator actions are not available to incorporate in the simulation.

At 3:00 am natural circulation cooldown was initiated by opening the steam dump and bypass valves. Pressurizer level was controlled after this point by means of charging and letdown. The temperature difference between the hot and cold legs continued at about 14°K (25°F), Figure 3. Shortly thereafter, the cooldown rate was reduced which leveled off the pressure decline at 14.3 mP (2080 psia), 3:32 am in Figure 1. Pressurizer level for the simulation was stabilized after this point at 32% of full

scale, Figure 2. Initiation of auxiliary spray at 3:45 am caused pressure to drop again until 4:00 am. A reduction in the cooldown rate at this time caused the pressure to level off at 12.1 mP (1750 psia). Spraying in the pressurizer was resumed at 4:30 am. The pressure began dropping at this time at a rate of about 3.4 mP/hr (500 psia/hr), a rate which was maintained until about 7:00 am. Pressurizer level began to rise at about 4:30 am, peaked near 40% of full scale at 5:00 am, and dropped back to 32% by 5:15 am. Loop temperature declined initially at a rate of 42°K (75°F) per hour. The average cooldown rate from 3:00 am to 6:00 am was 33°K (60°F) per hour. The cooldown rate declined further after this time. The loop temperature difference varied from 11 to 17°K (20 to 30°F) during this period. The simulation closely reproduces the measured loop temperatures, pressure and pressurizer level behavior from 3:03 to 6:00 am, as shown by Figures 1-3.

As the natural circulation cooldown progressed, the operators noted, at about 6:00 am, that the loop temperatures were approaching the entry conditions for shutdown cooling more rapidly than the pressurizer was being cooled down. At this time charging to the loop was secured and full charging flow was directed to auxiliary spray in the pressurizer to increase the pressurizer cooldown rate. Between 6:15 and 7:15 am the pressurizer level rose dramatically as shown in Figure 2. This was an unexpected result. Subsequently, a drop in pressurizer level was observed when charging flow was shifted from auxiliary spray to the cold legs. Further cycling of the charging pump alignment from auxiliary spray to the cold legs produced additional pressurizer level cycles, Figure 2.

Subsequent evaluations of the event led to the conclusion that a bubble was formed in the upper head of the reactor vessel.¹⁰ Stagnation of the upper head fluid produced a slower cooldown rate in the upper head than in the rest of the primary system. When the RCS pressure reached the saturation point in the upper head, fluid began flashing to form the steam bubble. Effectively, a second pressurizer was established in the upper head.

Auxiliary spray to the pressurizer was increased in the simulation at 5:30 am. Charging to the cold legs was secured at 6:00 am which further increased auxiliary spray to the pressurizer. The increased spray flow lowered the RCS pressure below the saturation pressure in the upper head. Flashing occurred

in the upper head which formed a bubble beginning at 7:00 am for the simulation, Figure 4. The hot upper head fluid was pushed into the RCS (where it mixed with the cooler RCS fluid) and forced RCS coolant into the pressurizer which raised the pressurizer level, Figure 2.

Realignment of charging flow to the cold legs and an increase of letdown flow at 7:07 am, for the simulation, ended the pressurizer level increase and initiated refill of the upper head. The surge of cooler RCS water into the isolated upper head steam region cooled the upper head. This induced a significant reduction in pressurizer level, Figure 2, thus completing the level oscillation cycle. Subsequent repetition of the sequence of charging and letdown actions in the simulation produced two more level oscillations, Figure 2.

At the end of the third level oscillation, the operators stopped letdown and started charging with three charging pumps in an apparent effort to eliminate the possible bubble in the system.¹⁰ Evaluations done after the event concluded that the upper head bubble probably collapsed as a result of this maneuver.¹⁰

These operator actions at the end of the third oscillation were simulated by closing the letdown valve, turning on two charging pumps, and later turning on a third charging pump as indicated in Table 1. The simulation shows that the upper head bubble collapses at 8:43 am. After collapse of the bubble, which is observed by the increase in pressurizer level, operator actions to reinstate cooldown were simulated by switching one charging pump to auxiliary sprays, turning off the other two charging pumps, and opening the letdown valve. Soon, as a result of the cooldown, a new bubble is generated in the upper head with a corresponding increase in pressurizer level. Once more, charging flow is switched from sprays to cold legs to stop the pressurizer level increase. At this point the simulation was terminated.

The relative magnitudes of the measured and simulated pressurizer level peak, after the first bubble is drawn in the upper head, differ due to two effects, Figure 2. The pressurizer level measurements are based on a calibration at hot, full power conditions. However, at the time the bubble is drawn, colder, more dense fluid enters the pressurizer. Recalibration for the actual fluid density would lower the magnitude of the measured level and improve

agreement of the simulation with data. Details of the charging and letdown rates, charging alignment, and flow rates used by the plant operators are not available. As a result, the values used in the simulation are approximate. Agreement might be improved by varying the timing and nature of the simulated operator actions.

The simulation reproduces the pressurizer and upper head level oscillation sequence observed in the plant transient. The ability to produce small or large level oscillations in the pressurizer as seen in the plant data is shown to be achievable in the simulation by proper manipulation of the charging pump alignment, charging flow rate and letdown flow rate. Refill of the upper head, which is believed to have occurred in the plant at 8:30 am¹⁰, is demonstrated in the simulation.

IV. UPPER HEAD VOIDING SIMULATION FOR OPERATOR TRAINING

On an NSSS where the upper guide tube plate separates the reactor vessel upper plenum and upper head, stagnation of fluid in the upper head complicates a natural circulation cooldown. Lowering RCS pressure for entry into shutdown cooling before the upper head is sufficiently cooled may produce voiding in the upper head. The voided upper head acts as a second pressurizer, but it lacks sprays or heaters for control of the pressure. Cooling the upper head by heat transfer to the containment or the upper plenum is slow and the condensate inventory may be exhausted before the pressure is low enough to use the shutdown cooling system. A more rapid upper head cooling procedure may be needed.

One possibility is to drain the hot water from the upper head and refill the upper head with cooler water from the RCS. Spraying in the pressurizer lowers system pressure below the saturation pressure in the upper head and flashing of the upper head fluid produces steam which forces the hot coolant out of the upper head. Possible mechanisms for refilling the upper head with cooler water are to charge to the cold legs, activate the pressurizer heaters, or open the reactor vessel head vent. Due to the complexities of the physical processes, operator training is needed to make effective use of these procedures.

Simulations of three procedures for refilling the upper head are shown to demonstrate the value of the C-E simulator for operator training. The simulation is based on the plant transient simulation discussed in

Section III. Each simulation begins after formation of the initial bubble in the upper head has raised the pressurizer level to 75%, at 7:07 am. For all cases, charging flow is switched from the pressurizer sprays to the cold legs and letdown flow is increased. After five minutes, at 7:12 am, the initiating sequence for each of the upper head refill procedures begins.

For the charging pump procedure, charging flow from three pumps is directed to the cold legs and letdown is stopped. This collapses the upper head bubble in 26 minutes, Figure 5. The operator can tell when the upper head is filled because the pressurizer level begins to rise as soon as the upper head is filled, Figure 6. System pressure also rises when the upper head is refilled providing another indication to the operator, Figure 7. Cooler fluid refills the upper head providing the desired cooling effect, Figure 8.

In the second procedure, activation of all of the pressurizer heaters swells the fluid volume in the pressurizer and forces RCS fluid back into the upper head. The upper head refills in 30 minutes, Figure 5. Pressurizer level again signals that the upper head is refilled, Figure 7. Addition of some charging flow to the cold legs along with activation of the heaters would avoid raising the system pressure. The upper head fluid is cooled to a lesser degree than with use of the charging pumps.

The upper head vent is fully opened in the third example. This refills the upper head in 11 minutes, Figure 5. Level in the pressurizer drops sharply here until the upper head is filled providing an indication of upper head filling. System pressure also drops as it would for a very small loss of coolant accident, Figure 7. Substantially cooler water is brought into the upper head, Figure 8.

The simplified training examples shown here were chosen to produce less complicated results for easier understanding. More complicated procedures combining the refill approaches or using multiple operator actions (as are believed to have occurred in the plant transient) can be simulated. Interactions between the steam region in the upper head and that in the pressurizer, effectively two pressurizers, can be simulated with all of the relevant driving forces. The sprays and the heaters control the pressurizer. The head vent and the loop charging flow influence the upper head behavior. Manipulation of letdown affects the total inventory. In addition, there is

another effect which is represented. When the pressure in either steam region reaches saturation, flashing begins which tends to push coolant out of that vessel. Thus the relative temperature of the two vessels controls which will reach saturation and flash first. As a result, changes of thermal states due to operator actions further complicates understanding of plant behavior. Use of a simulator with first principles models, such as the C-E full scope simulator, for operator training will improve operator understanding and performance for natural circulation transients.

The simulator has additional applications besides operator training. Alternative operating procedures can be compared to develop insight into their merits. Pitfalls in the procedures can be explored to prepare the operator to detect and avoid them. Examples include overfilling or packing the pressurizer, increasing system pressure as occurred with the heaters alone, or running out of condensate for feedwater before reaching shutdown cooling conditions. Examining the effect of alternative actions could be used to develop recommended approaches to handle natural circulation cooldowns.

V. CONCLUSIONS

The C-E simulator thermal-hydraulic models are validated for natural circulation transients. The models are able to reproduce initiation of natural circulation after pump trip, plant cooldown under single-phase RCS fluid flow conditions and formation of a steam bubble in the upper head of the reactor vessel. In addition, the nonequilibrium, two-phase fluid models reproduce level oscillations in the upper head and pressurizer and refill of the upper head in response to variations in the charging and letdown flows. The plant behavior observed in a C-E PWR is reproduced within the constraints imposed by available information about the operator actions.

There are several advantages to using a simulator with first principles models instead of the actual plant for reactor operator training. These include the ability to explore alternate means to cool the upper head during a natural circulation cooldown and exploration of alternatives or complications that would not be feasible with a power reactor. The fidelity of the C-E simulator is good enough that it could support these applications.

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TABLE 1

Sequence of Events for Natural Circulation Transient

<u>Time</u>	<u>Event</u>	<u>Time</u>	<u>Event</u>
2:33 am ^a	Reactor trip from 99.6% power. Turbine/Generator trip.	5:13	Increase charging. Decrease sprays.
2:35 ^a	Trip all Reactor Coolant Pumps (RCP).	5:30	Increase sprays.
2:38 ^a	Restart one RCP.	6:00	End loop charging to increase sprays. Nominal letdown.
2:39 ^a	Trip RCP.	6:40 ^a	Pressurizer level increases.
2:43 ^a	Natural circulation begins. Small changes omitted due to limited information.	7:00	Bubble forms in upper head. Natural circulation uninterrupted by upper head bubble.
2:50 ^a (approx.)	Pressure, pressurizer level and loop temperature changes after initial transient are damped.	7:07 ^b	Charging switched to cold legs. Letdown increased.
3:00 ^a	Open steam dump and bypass to start natural circulation cooldown. Charging and letdown used to control pressurizer level, pressurizer heaters off.	7:28 ^c	Charging switched to sprays. Letdown reduced.
3:20	Reduce cooldown rate, turn on pressurizer heaters.	7:37 ^b	Charging switched to cold legs. Letdown increased.
3:26	Turn off heaters, decrease charging.	7:42 ^c	Charging switched to sprays. Letdown reduced.
3:40	Increase cooldown rate. Increase charging.	8:01 ^b	Charging switched to cold legs. Letdown increased.
3:46	Open auxiliary spray line. Increase cooldown rate.	8:18 ^a (approx.)	Increase cold leg charging to two pumps. Letdown terminated.
4:00	Reduce cooldown rate. Decrease charging, auxiliary sprays off.	8:35 ^a (approx.)	Increase cold leg charging to three pumps.
4:30	Increase cooldown rate. Increase charging, auxiliary sprays on.	8:43	Upper head fills solid.
5:00	Reduce charging and letdown. Cooldown rate reduced.	8:49	Put one charging pump to sprays, rest off. Nominal letdown.
		9:25 ^c	Charging flow switched to sprays. Open letdown.
		9:35	End simulation.

a) From sequence of events in Reference 10.

b) End pressurizer refill and begin refill of upper head.

c) Resume cooldown and pressurizer refill.

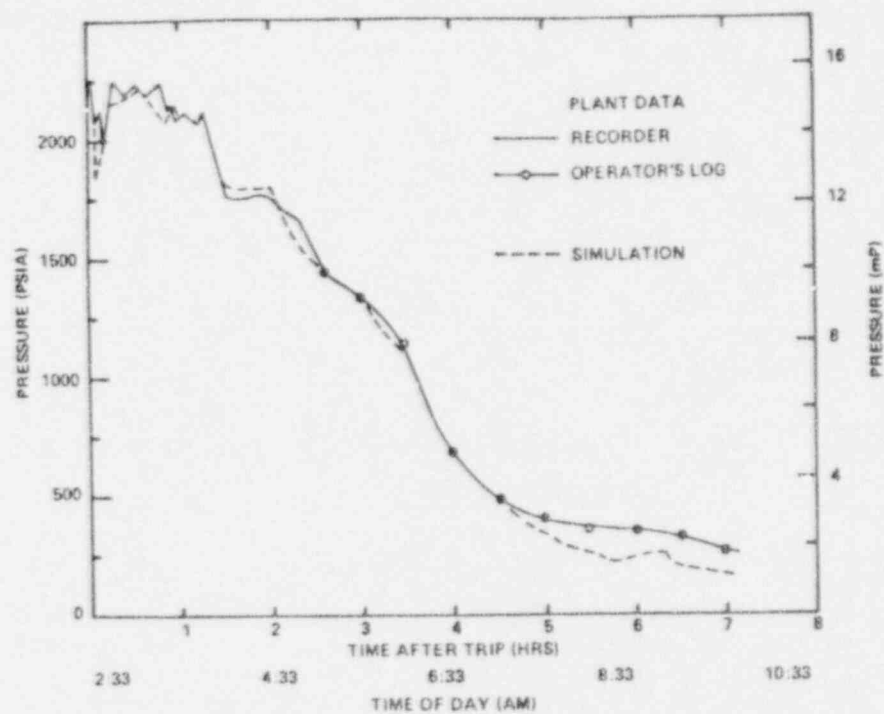


FIGURE 1 PLANT TRANSIENT PRESSURE

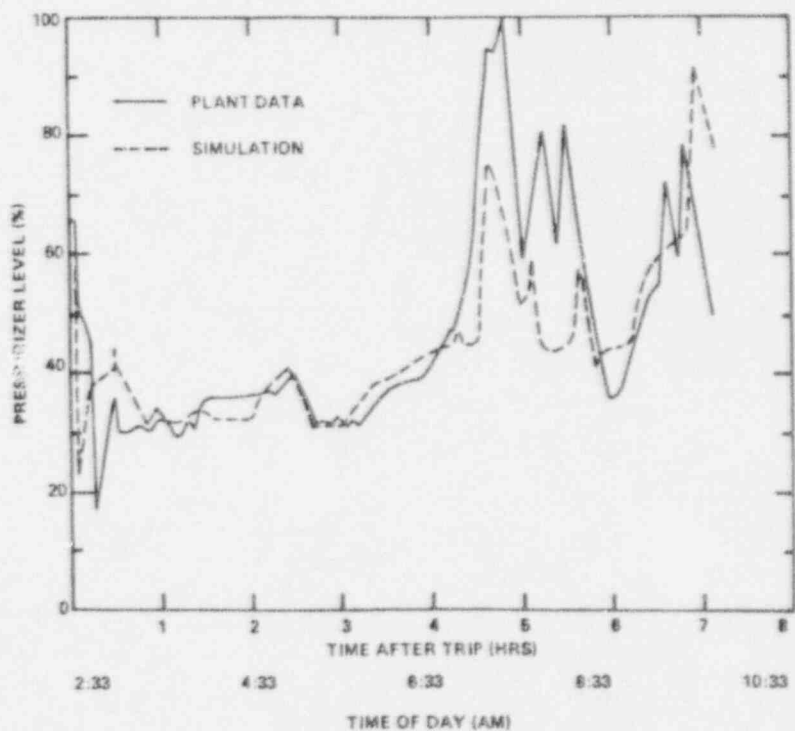


FIGURE 2 PLANT TRANSIENT PRESSURIZER LEVEL

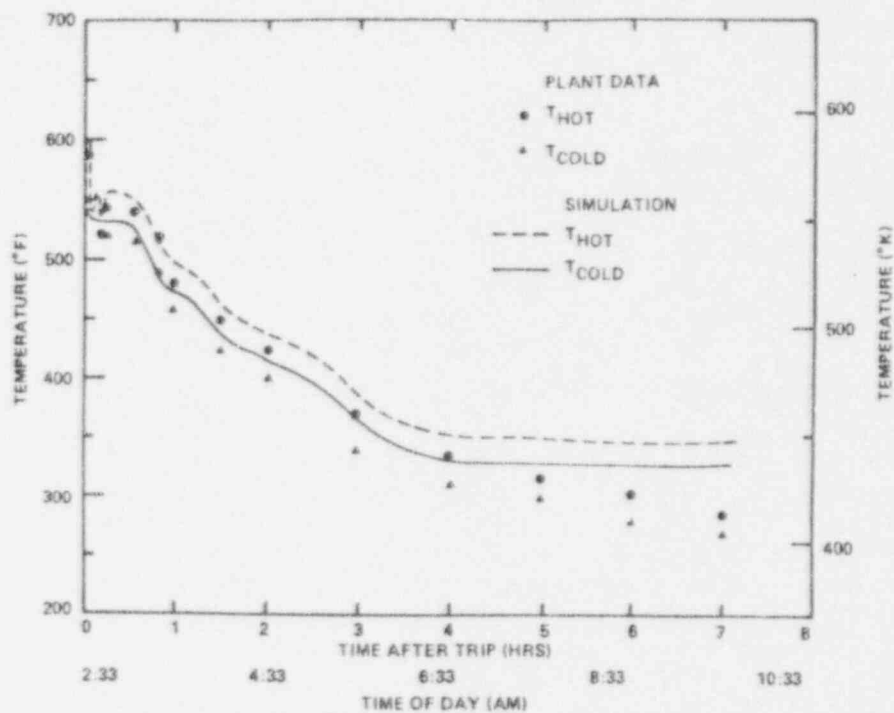


FIGURE 3 PLANT TRANSIENT LOOP TEMPERATURES

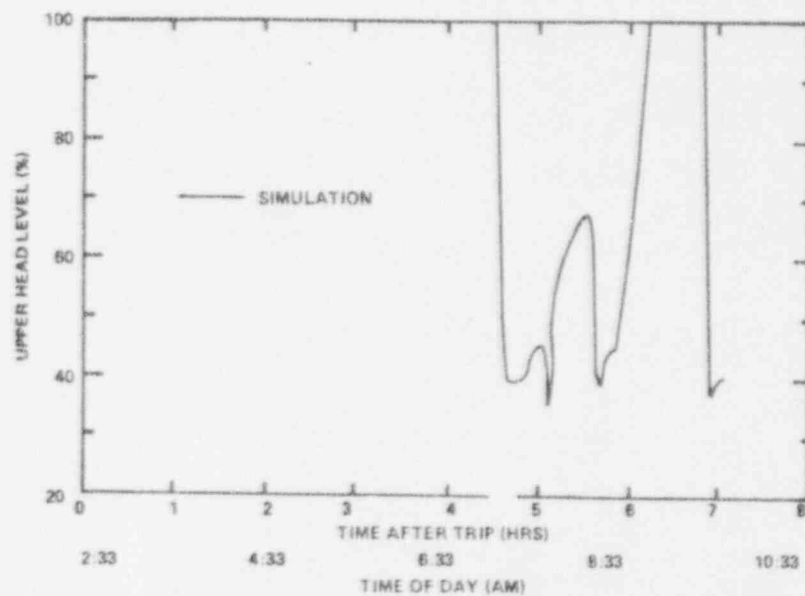


FIGURE 4 PLANT TRANSIENT UPPER HEAD LEVEL

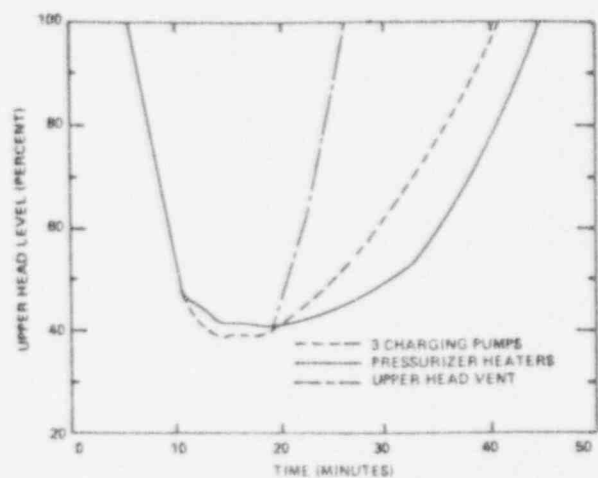


FIGURE 5 UPPER HEAD LEVEL

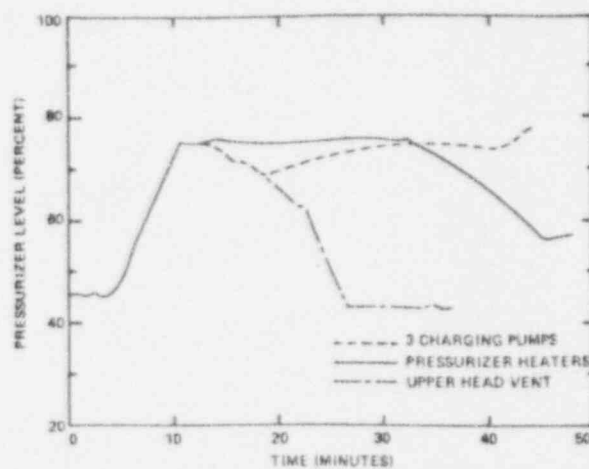


FIGURE 6 PRESSURIZER LEVEL

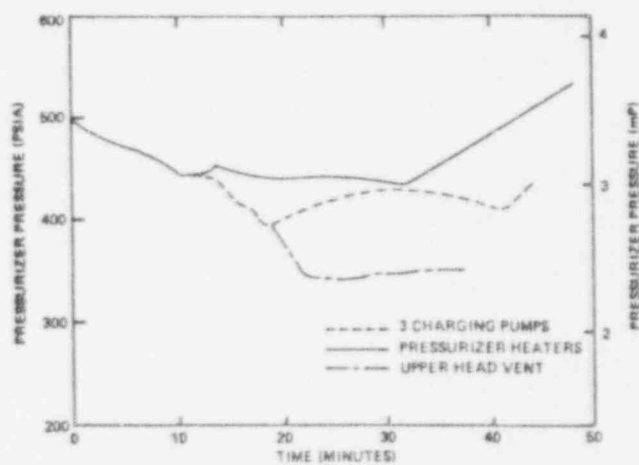


FIGURE 7 PRESSURIZER PRESSURE

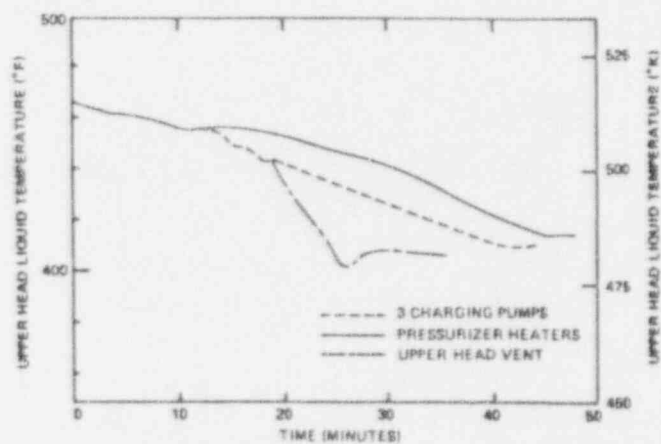


FIGURE 8 UPPER HEAD LIQUID TEMPERATURE