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REACTOR

4.1 SUMMARY DESCRIPTION

The reactor assembly for the Duane Arnold Energy Center (DAEC) consists of the reactor vessel, its internal components of the core, steam separator and dryer assemblies, and jet pumps. Also included in this assembly are the control rods, control rod drive (CRD) housings, and the control rod drives.

4.1.1 REACTOR VESSEL

The reactor vessel design is described in Chapter 5.

4.1.2 REACTOR INTERNAL COMPONENTS

The major reactor internal components are the core (fuel, channels, control blades, and incore instrumentation), core support structure (including the shroud, top guide, and core plate), shroud head and steam separator assembly, steam dryer assembly, feedwater spargers, core spray spargers, and jet pumps. Except for the Zircaloy in the reactor core, these reactor internals are made of stainless steel or other corrosion-resistant alloys. Of the preceding components, the fuel assemblies (including fuel rods and channel), control blades, incore instrumentation, shroud head and steam separator assembly, and steam dryers are removable when the reactor vessel is open.

4.1.2.1 Reactor Core

4.1.2.1.1 General

The design of the boiling-water reactor (BWR) core (including fuel) is based on the proper combination of many design variables and operating experience. This contributes to the achievement of high reliability.

A number of important features of the BWR core design are summarized below:

1. The BWR core mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The moderate pressure level characteristic of a direct-cycle reactor (approximately 1000 psia) results in moderate cladding temperatures and stress levels.

2. The low coolant saturation temperature, high heat-transfer coefficients, and neutral water chemistry of the BWR are significant, advantageous factors in minimizing zirconium alloy temperature and associated temperature-dependent corrosion and hydride buildup.

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   The relatively uniform fuel cladding temperatures throughout the core minimize the migration of the hydrides to cold cladding zones and reduce thermal stresses.

3. The basic thermal and mechanical criteria applied in the design have been proved by the irradiation of statistically significant quantities of fuel. The design heat-transfer rates and linear heat generation rates (LHGRs) are similar to values proved in fuel assembly irradiation.

4. The design power distribution used in sizing the core represents a worst expected state of operation.

5. The General Electric thermal analysis basis, GETAB, is applied to ensure that more than 99.9% of the fuel rods in the core are expected to avoid boiling transition for the most severe abnormal operational transient described in Chapter 15. The possibility of boiling transition occurring during normal reactor operation is insignificant.

6. Because of the large negative moderator density coefficient of reactivity, the BWR has a number of inherent advantages. These are the use of coolant flow for load following, inherent self-flattening of the radial power distribution, ease of control, spatial xenon stability, and the ability to override xenon in order to follow load.

BWRs do not have instability problems due to xenon. This has been demonstrated by special tests that have been conducted on operating BWRs in an attempt to force the reactor into xenon instability and by calculations. No xenon instabilities have ever been observed in the test results. All of the indicators show that xenon transients are highly damped in a BWR because of the large negative power coefficient of reactivity.

Important features of the reactor core arrangement are as follows:

1. The bottom-entry cruciform control rods consist of boron carbide in stainless steel tubes surrounded by a stainless steel sheath. Rods of this design have accumulated thousands of hours of service in operating BWRs without significant failure. Beginning with Cycle 9 operation, some replacement rods utilize a new design which increases both the mechanical and nuclear lifetimes. This is the Hybrid I Control Rod Assembly also called the GE Duralife D160 design. Beginning with Cycle 11, some controls rods were replaced with the G.E. Duralife D-230 control rod assemblies. Beginning with Cycle 20, some control rods were
2012-020 replaced with GE Marathon control rod assemblies. Beginning with Cycle 24, some control rods were replaced with Westinghouse CR-99 control rod assemblies. These assemblies are described in Section 4.6.1.2.5.

2. The fixed incore fission chambers provide continuous power range neutron flux monitoring. A guide tube in incore assemblies provides for a traversing ion chamber for calibration and axial detail. One incore assembly does not have a guide tube for a traversing ion chamber. Modeling techniques in the three dimensional core simulator are used to provide information for calibration and axial detail. This assembly was modified to accommodate monitoring equipment for Noble metals injection and coating. Source and intermediate range detectors are located in the core and are axially retractable. The incore location of the startup and source range instruments provides the coverage of the large reactor core and provides an acceptable signal-to-noise ratio and neutron-to-gamma ratio. All incore instrument leads enter from the bottom and the instruments are in service during refueling. Incore instrumentation is further discussed in Section 7.7.

3. Experience has shown that the operator, using the incore flux monitoring system, can maintain the desired power distribution within a large core by proper control rod scheduling.

2012-020 4. The zirconium alloy channels provide a fixed flow path for the boiling coolant, serve as a guiding surface for the control rods, and protect the fuel during handling operations.

5. Mechanical reactivity control permits criticality checks during refueling and provides maximum plant safety. The core is designed to be subcritical at any time in its operating history, with any one control rod fully withdrawn.

6. The selected control rod pitch represents a practical value of individual control rod reactivity worth, and allows adequate clearance below the pressure vessel, between CRD mechanisms, for ease of maintenance and removal.

4.1.2.1.2 Core Configuration

The reactor core is arranged as an upright circular cylinder containing a large number of fuel cells and is located in the reactor vessel. The coolant flows upward through the core.

4.1.2.1.3 Fuel Assembly Description

The BWR core is composed essentially of two components: fuel assemblies and control rods. The fuel assembly and control rod-mechanical configurations are basically the same as used [redacted] in all subsequent GE BWRs. Further discussion is contained in Sections 4.2 and 4.3.

4.1.2.1.3.1 <u>Fuel Rod</u>

2012-020 A fuel rod consists of uranium dioxide pellets and a zirconium alloy cladding tube. For the barrier fuel design, the cladding consists of the same zirconium alloy base material with the innermost part of the cladding replaced by a thin zirconium liner. This liner is mechanically bonded to the base zircaloy material during manufacture. The purpose of the zirconium liner is to improve stress resistance in pellet clad interaction. A fuel rod is made by stacking pellets into the cladding tube, which is evacuated and backfilled with helium and sealed by welding zircaloy end plugs in each end of the tube. The ASME Boiler and Pressure Vessel Code, Section III, is used as a guide in the mechanical design and stress analysis of the fuel rod. The fuel rod is designed to withstand applied loads, both external and internal. The fuel pellet is sized to provide sufficient clearance within the cladding tube and to accommodate axial and radial differential expansion between fuel and clad. Overall fuel rod design is conservative in its accommodation of the mechanisms affecting fuel in a BWR environment.

4.1.2.1.3.2 <u>Fuel Bundle</u>

Each fuel bundle contains fuel rods and water rods, which are spaced and supported in a square array (<u>e.g.</u>, 8 x 8, 10 x 10, etc.) by spacers and a lower and upper tie plate. The fuel bundle has two important design features as follows:

1. Each fuel rod is free to expand in the axial direction.

2. The structural design permits the removal and replacement of individual fuel rods if required.

The fuel assemblies, of which the core is comprised, are designed to meet all criteria for core performance and to provide ease of handling. Selected fuel rods in each assembly differ from the others in uranium enrichment. This arrangement produces more uniform power production across the fuel assembly, and thus allows a significant reduction in the amount of heat-transfer surface required to satisfy the design thermal limitations. Further discussion may be found in Section 4.2.

4.1.2.1.4 Fuel Assembly Support and Control Rod Location

All peripheral fuel assemblies and their individual peripheral fuel support pieces are supported by the core plate. Otherwise, individual fuel assemblies in the core rest on fuel support pieces mounted on top of the control rod guide tubes. Each guide tube, with its fuel support piece, bears the weight of four assemblies and is supported by a CRD penetration nozzle in the bottom head of the reactor vessel. The core plate provides lateral support and guidance at the top of each control rod guide tube.

The top guide, mounted inside the shroud, provides lateral support and guidance for each fuel assembly. The reactivity of the core is controlled by cruciform-shaped control rods containing boron carbide and their associated mechanical-hydraulic drive system. The control rods occupy alternate spaces between fuel assemblies. Each independent control rod drive enters the core from the bottom, accurately positions its associated control rod during normal operation, and yet exerts approximately 10 times the force of gravity to insert the control rod during scram mode of operation. Bottom entry allows optimum power shaping in the core, ease of refueling, and convenient CRD maintenance.

4.1.2.2 <u>Shroud</u>

Information on the shroud is contained in Section 3.9.5.

4.1.2.3 <u>Shroud Head and Steam Separator Assembly</u>

Information on the shroud head and steam separators is contained in Section 3.9.5.

4.1.2.4 <u>Steam Dryer Assembly</u>

Information on the steam dryer assembly is contained in Section 3.9.5.

4.1.3 REACTIVITY CONTROL SYSTEMS

4.1.3.1 Operation

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during the operation of the reactor by the manipulation of selected patterns of rods. The rods, which enter from the bottom of the near-cylindrical reactor core, are positioned to counterbalance steam voids in the top of the core and effect significant power flattening. These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods in the control system.

The reactivity control function requires that all rods be available for either reactor scram or reactivity regulation. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, hydraulically actuated drive mechanisms that allow either axial positioning for reactivity regulation or rapid scram insertion. The design of the rod-to-drive connection permits each blade to be attached or detached from its drive, without disturbing the remainder of the control system. The bottom-mounted drives permit the entire control system to be left intact and operable for tests with the reactor vessel open.

4.1.3.2 Description of Control Rods

The physical description of control rods is provided in Section 4.6.1.2.5.

The control rods can be positioned at 6-in. steps and have a nominal withdrawal and insertion speed of 3 in./sec. The velocity limiter is a device which is an integral part of the control rod and protects against the low probability of a control rod drop accident. It is designed to limit the free-fall velocity and consequent reactivity insertion rate of control rods, so that minimal fuel damage would occur. It is a one-directional device so that control rod scram time is not significantly affected.

The control rods are cooled by the core leakage (bypass) flow. As shown in Figure 4.1-1, the core leakage flow is made up of recirculation flow that leaks through several leakage flow paths, the most important of which are as follows:

1. The area between the fuel channel and the fuel assembly lower tie plate.

2. Holes in the lower tie plate.

3. The area between the fuel assembly lower tie plate and the fuel support piece.

4. The area between the fuel support piece and the control rod guide tube.

5. The area between the control rod guide tube and the core support plate.

6. The area between the core support plate and the shroud.

4.1.3.3 Supplementary Reactivity Control

The core control requirements are met by the use of the combined effects of the movable control rods, supplementary burnable poison, and the variation of reactor coolant flow. The supplementary burnable poison is gadolinia (Gd_2O_3) mixed with the uranium dioxide fuel in selected fuel rods in each fuel bundle.

4.1.4 NUCLEAR DESIGN

The nuclear evaluations are performed using the analytical tools and methods described in Section 4.3. The nuclear evaluation procedure is best described in two parts: lattice analysis and core analysis.

Most of the lattice analysis is performed during the bundle design process using the methods in Section 4.3. The results of these single-bundle calculations are reduced to "libraries" of lattice reactivities, relative rod powers, and the cross-sections averaged over a reduced number of energy groups as functions of instantaneous void, exposure, exposure-void history, control state, and fuel and moderator temperature for use in the core analysis. These analyses are dependent on fuel lattice parameters only and are, therefore, valid for all operating plants and cycles to which they are applied.

The core analysis is unique for each reload. It is performed in the months preceding the reload to demonstrate that the core meets all applicable safety limits. The principal tool used in the core analysis is the three-dimensional BWR simulator code, which computes power distributions, exposure, and reactor thermal-hydraulic characteristics, with spatially varying voids, control rods, burnable poisons, and other variables.

4.1.5 THERMAL-HYDRAULIC DESIGN

The core thermal-hydraulic analysis and design are discussed in Section 4.4.

[redacted]

4.2 FUEL SYSTEM DESIGN

The format of this section corresponds to Standard Review Plan 4.2 in NUREG - 0800. Most of the information presented will be by reference to the approved General Electric (GE) report.[1]

4.2.1 DESIGN BASES

The fuel assembly must be designed to ensure that possible fuel damage would not result in the release of radioactive materials in excess of applicable regulations. The adequacy of the fuel assembly is demonstrated if it is shown to provide substantial fission product retention capability during all potential operational modes and sufficient structural integrity to prevent operational impairment of any reactor safety equipment. The fuel assembly and its components are designed to withstand the loadings documented in Section 2.2 of Reference 1. Specific criteria and limits that ensure that these bases are met are given below. In addition, for advanced fuel designs, specific licensing criteria are applied (Reference 15).

4.2.1.1 <u>Fuel System Damage Limits</u>

4.2.1.1.1 Stress-Strain Limits

Stress and equivalent strain limits for normal and abnormal operational transient loads on the fuel rod and other bundle component analyses are documented in Subsection 2.2.1.1 of Reference 1.

4.2.1.1.2 Fatigue Limits

Stress/cycle limits for fatigue analyses are given in Subsection 2.2.1.2 of Reference 1.

4.2.1.1.3 Fretting Wear Limits

Per Section 2.2.1.3 of Reference 1, the fuel assembly is evaluated to ensure that fuel will not fail due to fretting wear of the assembly components

4.2.1.1.4 Oxidation, Hydriding, and Corrosion Limits

There are design limits for cladding oxidation, hydriding, and corrosion.
2012-020 Oxidation and corrosion are considered in the mechanical and thermal-mechanical design analyses. Criteria for these analyses are given in Subsection 2.2.1.4 of Reference 1. Hydriding is controlled through the use of a specification limit of the amount of hydrogen
2012-020 permitted in a manufactured fuel rod. In addition, an upper bound corrosion thickness limit and an upper bound cladding hydrogen content limit as discussed in Subsection 2.2.2.7.2 of Reference 1 are applied to prevent significant localized cladding ductility loss.

4.2.1.1.5 Dimensional Change Limits

Rod-to-rod and rod-to-channel deflection limits are given in Subsection 2.2.1.5 of Reference 1. Manufacturing tolerances, axial load and thermal effects are included in the fuel design and thermal and mechanical analyses. Maximum allowable control blade-to-channel clearance is given in Reference 2.

4.2.1.1.6 Internal Gas Pressure Limits

The internal gas pressure within the fuel rod is limited so that the cladding creepout rate is not expected to exceed the instantaneous fuel swelling rate. The internal gas pressure as a function of exposure is an input to the fuel rod mechanical design analyses. Criteria for these analyses are given in Subsection 2.2.1.6 of Reference 1.

4.2.1.1.7 Hydraulic Load Limits

The fuel assembly is evaluated to ensure that vertical liftoff forces are not sufficient to unseat the bundle to a degree that the bundle could interfere with control blade insertion. Normal operational hydraulic loads are conservatively bounded by the combined loss-of-coolant accident (LOCA) plus safe shutdown earthquake (SSE) loading. Design limits for this faulted condition are given in Subsection 2.2.1.7 of Reference 1.

4.2.1.1.8 Control Rod Reactivity Limits

Sections 4.3.1 and 4.3.2.4 provide the control rod reactivity basis. An evaluation of control blade lifetime as provided by nuclear and mechanical lifetime limits is given in
2012-020 References 3 and 21.

4.2.1.2 <u>Fuel Rod Failure Limits</u>

4.2.1.2.1 Hydriding Limits

Hydriding limits are discussed in Section 4.2.1.1.4.

4.2.1.2.2 Cladding Collapse Limits

If axial gaps in the fuel pellet column occur from densification, the cladding has the potential of collapsing into a gap. To preclude collapse, the criterion provided in Section 8 of Reference 4 is met.

4.2.1.2.3 Fretting Wear Limits

These limits are addressed in Section 4.2.1.1.3

4.2.1.2.4 Overheating of Cladding Limits

The fuel cladding integrity safety limit for the minimum critical power ratio (MCPR) (Subsection 2.2.2.4 and Subsection 4.3.1 of Reference 1) ensures that overheating of the cladding does not occur.

4.2.1.2.5 Overheating of Pellet Limits

2012-020 No fuel centermelt occurs for normal operation and anticipated operational occurrences as documented in Subsection 2.2.2.5 of Reference 1.

4.2.1.2.6 Excessive Fuel Enthalpy Limits

Clad failure threshold is 170 cal/g. This limit is identified in Amendment 7 to Reference 1.

4.2.1.2.7 Pellet-Cladding Interaction Limits

The fuel rods are evaluated to ensure that fuel rod failure due to pellet-clad
2012-020 mechanical interaction will not occur, i.e. the 1% strain criterion is not exceeded. These limits are given in Subsection 2.2.2.7 of Reference 1.

4.2.1.2.8 Bursting Limits

Section 1.B of Appendix K to 10 CFR 50 specifies that each LOCA evaluation model shall include a provision for predicting cladding swelling and rupture due to axial temperature distribution and differential pressure between inside and outside cladding, (Ref. NUREG-0630). This specification is met as documented in Volume II of Reference 6.

4.2.1.2.9 Mechanical Fracturing Limits

Mechanical breaking under normal and abnormal operational transients is bounded by the combined LOCA plus SSE loading. Limits for this faulted condition are given in Subsection 2.2.2.9 of Reference 1.

4.2.1.3 <u>Fuel Coolability Limits</u>

4.2.1.3.1 Cladding Embrittlement Limits

Peak clad temperature and maximum cladding oxidation limits documented in 10 CFR 50.46 ensure that cladding embrittlement does not occur. Conformance to these limits is given in Volume III of Reference 6.

4.2.1.3.2 Violent Expulsion of Fuel Limits

The limit for severe reactivity accidents is 280 cal/g. This limit is documented in Amendment 7 to Reference 1.

4.2.1.3.3 Generalized Cladding Melt Limits

As documented in the Standard Review Plan, the generalized cladding melt limit is bounded by the cladding embrittlement limit given in Section 4.2.1.3.1.

4.2.1.3.4 Fuel Rod Ballooning Limits

Criteria for fuel rod ballooning are given in Section 4.2.1.2.8.

4.2.1.3.5 Structural Deformation Limits

Faulted limits for the DBE plus LOCA analysis are given in Section 4.2.1.2.9.

4.2.2 DESCRIPTION AND DESIGN DRAWINGS

The fuel assembly consists of a fuel bundle and a channel that surrounds it. The DAEC has an evolution of fuel designs in the core. The reference core-loading pattern for each cycle of operation is documented in Section 4.3. A description and drawings of each fuel type and other bundle components for the current core are given in Reference 14. The channel is described and analyzed in Reference 2. The reactivity control assembly is described in Section 4.6.

4.2.3 DESIGN EVALUATION

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The following sections document that the design bases described in Section 4.2.1 are met. Methods used to demonstrate this compliance are operating experience, prototype testing, and analytical predictions. Most of these methods are documented in other approved reports, which are referenced in the following sections. As stated in Section 4.2.1, specific licensing criteria are applied to advanced fuel designs (Ref. 15). For the GE12 fuel design introduced during Cycle 17, this compliance is demonstrated in Reference 16. For the GE14 fuel design introduced during Cycle 18, this compliance is demonstrated in Reference 18. For GNF2 fuel design introduced during Cycle 24 and for GNF2.02 fuel design introduced during Cycle 27, this compliance is demonstrated in Reference 20. Note that GNF2.02 fuel design is identical to the GNF2 fuel design, with two improved mechanical components: a modified spacer and the Defender PLUS filter cartridge; thus GNF2.02 is an improved but equivalent version of the GNF2 fuel design.

4.2.3.1 Fuel System Damage Evaluation

4.2.3.1.1 Stress-Strain Evaluation

Analyses of the mechanical integrity of the fuel bundle demonstrate that the design bases are met. Analytical methods and results are documented in Section 2.2.1.1 of Reference 1.

4.2.3.1.2 Fatigue Evaluation

The analysis of the cumulative fatigue damage on key components shows that the design bases are met. Analytical methods and results are documented in Section 2.2.1.2 of Reference 1.

4.2.3.1.3 Fretting Wear Evaluation

Extensive out-of-reactor and in-reactor testing and surveillance verify that fretting wear is not a problem with the current fuel design. Details of the tests are given in Section 2.2.1.3 of Reference 1. Additional information is given in Section VII of Reference 8.

4.2.3.1.4 Oxidation, Hydriding, and Corrosion Evaluation

Oxidation, Hydriding and Corrosion Products are discussed in Section 2.2.1.4 of Reference 1.

4.2.3.1.5 Dimensional Change Evaluation

Fuel rod deflection analysis results show that the design bases are met. Analytical methods and results are provided in Subsection 2.2.1.5 of Reference 1. Fuel rod bowing is addressed separately in Reference 9. Channel deflection analysis is given in Reference 2.

4.2.3.1.6 Internal Gas Pressure Evaluation

Results of the analysis for end-of-life internal gas pressure is provided in Section 2.2.1.6 of Reference 1.

4.2.3.1.7 Hydraulic Load Evaluation

Fuel rod vibration analysis is provided in Section 2.2.1.7 of Reference 1. A conservative evaluation showing there is no potential for fuel bundle lift is provided in Chapter 15.3.5.

4.2.3.1.8 Control Rod Reactivity Evaluation

The control rod reactivity evaluation is documented in Section 4.3.

4.2.3.2 Fuel Rod Failure Evaluation

4.2.3.2.1 Hydriding Evaluation

The evaluation of hydriding is given in Section 4.2.3.1.4

4.2.3.2.2 Cladding Collapse Evaluation

Cladding collapse is not calculated to occur in General Electric fuels documented in Section 2.2.2.2 of Reference 1.

4.2.3.2.3 Fretting Wear Evaluation

The evaluation of fretting wear is given in Section 4.2.3.1.3.

4.2.3.2.4 Overheating of Cladding Evaluation

Subsection 4.3.1 of Reference 1 describes the basis for the operating MCPR limit calculation. This cycle-dependent limit ensures that the MCPR fuel cladding integrity safety limit is not exceeded during abnormal operational transients.

4.2.3.2.5 Pellet Overheating Evaluation

Per Section 2.2.2.5.1 of Reference 1, the fuel rod is evaluated to ensure that fuel melting during normal steady state operation and whole core anticipated operational occurrences is not expected to occur. For local anticipated operational occurrences such as the rod withdrawal error, a small amount of calculated fuel pellet center melting may occur but is limited by 1% cladding circumferential plastic strain criterion.

4.2.3.2.6 Excessive Fuel Enthalpy Evaluation

The evaluation of reactivity events is given in Chapter 15.

4.2.3.2.7 Pellet-Cladding Interaction Evaluation

Transient analyses documented in Chapter 15 do not exceed the design basis of 1% plastic strain or MCPR fuel cladding integrity safety limits.

4.2.3.2.8 Bursting Evaluation

In accordance with Appendix K to 10 CFR 50, temperature and time of rupture at key exposures are given in the plant specific ECCS analysis in Chapter 15.2.

4.2.3.2.9 Mechanical Fracturing Evaluation

Per Section 2.2.2.9 of Reference 1, the fuel assembly is evaluated under Safety Shutdown Earthquake and LOCA loading conditions to ensure that loss of fuel assembly coolability, and interference to the degree that control blade insertion is prevented, will not occur.

4.2.3.3 Fuel Coolability Evaluation

4.2.3.3.1 Cladding Embrittlement Evaluation

Precluding cladding embrittlement is demonstrated in the LOCA analysis provided in Section 15.2.

4.2.3.3.2 Violent Expulsion of Fuel Evaluation

The DAEC has adapted a control rod withdrawal sequence which has been shown statistically to result in rod worths below those needed to exceed the peak fuel enthalpy limit of 280 cal/gm for fuel expulsion.

4.2.3.3.3 Generalized Cladding Melt Evaluation

The LOCA analysis documented in Chapter 15.2 is bounding for this evaluation.

4.2.3.3.4 Fuel Rod Ballooning Evaluation

Fuel rod ballooning is addressed in Section 4.2.3.2.8.

4.2.3.3.5 Structural Deformation Evaluation

Results of the SSE plus LOCA analysis are documented in Subsection 2.2.2.9 of Reference 1.

4.2.4 TESTING, INSPECTION, AND SURVEILLANCE PLANS

Fuel assembly testing and inspection are documented in Section 2.3.1 of Reference 1.

REFERENCES FOR SECTION 4.2

1. General Electric Company, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest NRC approved revision).

2. General Electric Company, BWR Fuel Channel Mechanical Design and Deflection, NEDE-21354-P (proprietary) and NEDO-21354, 1976.

3. General Electric Company, GE BWR Control Rod Lifetime, NEDE-30931-4-P, Rev. 4, May, 1996.

4. General Electric Company, Creep Collapse Analysis of BWR Fuel Using SAFE-COLAPS Model, NEDE-20606-P-A (proprietary) and NEDO-20606-A, 1976.

5. C. J. Paone, et al., Rod Drop Accident Analysis for Large BWR's, NEDO-10527, 1972.

6. General Electric Company, The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, NEDE-23785-P (proprietary), January 1985.

7. R. H. Buchholz (General Electric) letter MFN-097-81 to L. S. Rubsenstein (NRC), General Electric Fuel Clad Swelling and Rupture Model, May 15, 1981.

8. General Electric Company, 8 x 8 Fuel Development Support, NEDO-20377, 1975.

9. General Electric Company, Assessment of Fuel Rod Bowing in GE BWR's, NEDE-24284-P, 1980.

10. Deleted.

11. Deleted.

12. General Electric Company, Experience with BWR Fuel Through January 1981, NEDE-24343-P, 1976.

13. General Electric Company, Safety Evaluation of the General Electric Hybrid I Control Rod Assembly, NEDE-22290-A, September 1983.

14. General Electric Company, GE Fuel Bundle Designs, NEDE-31152P, (latest NRC approved version).

15. General Electric Company, Licensing Criteria for Fuel Designs (Amendment 22 to NEDE-24011-P-A and Corresponding NRC Staff Safety Evaluation), NEDO-31908, January 1991.

16. General Electric Company, GE12 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR-II), NEDE-32417P, December 1994.

17. Deleted.

18. General Electric Company, GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR-II), NEDC-32868P, December 1998.

19. Deleted.

2019-001 20. Global Nuclear Fuel, GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR-II), NEDC-33270P, Revision 7, October 2016.

2012-020 21. Westinghouse BWR Control Rod CR 99 Licensing Report, WCAP-16182-P-A, Revision 0, March 2005.

4.3 NUCLEAR DESIGN

The nuclear core design presented here is based on the D lattice fuel documented in Sections 1 and 2 of Reference 1. Specific core-loading patterns for the DAEC are given in the references in Table 4.3-1.

4.3.1 DESIGN BASES

The nuclear design bases are divided into two specific categories: (1) the safety design bases are those that are required for the plant to operate from safety considerations and (2) the plant performance design bases are those that are required to meet the objective of producing power in an efficient manner.

4.3.1.1 Safety Design Bases

The safety design bases are requirements that protect the nuclear fuel from a violation of the design integrity limits. In general, the safety bases fall into two categories: (1) the reactivity basis, which prevents an uncontrolled positive reactivity excursion, and (2) the overpower bases, which prevent the core from operating beyond the fuel integrity limits.

4.3.1.1.1 Reactivity Basis

The nuclear design shall meet the following basis: The core shall be capable of being rendered subcritical at any time or at any core conditions with the highest worth control rod fully withdrawn.

4.3.1.1.2 Overpower Bases

The limits on LHGR, MCPR, and the maximum average planar linear heat generation rate (MAPLHGR) as presented in the Core Operating Limits Report shall not be exceeded during steady-state operation.

4.3.1.2. Plant Performance Design Bases

The nuclear design shall meet the following bases:

1. The design shall provide adequate hot excess reactivity to attain the desired cycle length.

2. The design shall be capable of operating at rated conditions without exceeding the Technical Specification limits.

3. The nuclear design and reactivity control system shall allow continuous, stable regulation of reactivity.

4. The nuclear design shall have adequate reactivity feedback to facilitate normal operation.

Core nuclear design analyses results are used as inputs to the core transient and stability analyses and do not have separate limits.

4.3.2 DESCRIPTION

The BWR core design uses a light-water-moderated reactor, fueled with slightly enriched uranium dioxide. The use of water as a moderator produces a neutron energy spectrum in which fissions are caused principally by thermal neutrons. At normal operating conditions, the moderator boils, producing a spatially variable distribution of steam voids in the core. The BWR design provides a system for which reactivity changes are inversely proportional to the steam void content in the moderator. This void feedback effect is one of the inherent safety features of the BWR system. Any system input that increases reactor power, either in a local or gross sense, produces additional steam voids that reduce reactivity and thereby reduce power.

4.3.2.1 Nuclear Design Description

The reference core-loading pattern is the basis for all fuel licensing. The bundle and lattice designations for D lattice fuel enrichments are given in Reference 3. Uranium dioxide and gadolinia distributions for each bundle enrichment and typical lattice nuclear characteristics are given in Reference 3. In addition, specific licensing criteria are applied to advanced fuel designs (Ref. 4). For the GE12 fuel design introduced during
2012-020 Cycle 17, this compliance is demonstrated in Reference 5. For the GE14 fuel design
introduced during Cycle 18, this compliance is demonstrated in Reference 6. For GNF2
2019-001 fuel design introduced during Cycle 24 and for GNF2.02 fuel design introduced during
Cycle 27, this compliance is demonstrated in Reference 7. Steady-state core characteristics are also discussed. The reference core-loading pattern for each reload cycle is given in the references listed in Table 4.3-1.

4.3.2.2 Power Distribution

The core power distribution is a function of fuel bundle design, core loading, control rod pattern, core exposure distributions, and core coolant flow rate. The thermal performance parameters, MAPLHGR and MCPR (defined in Table 3-1 of Reference 1) limit unacceptable core power distribution.

4.3.2.3 Reactivity Coefficients

Reactivity coefficients are discussed in Section 3.2.3 of Reference 1.

4.3.2.4 Control Requirements

The nuclear design, in conjunction with the reactivity control system, provides an inherently stable system for BWRs.

The control rod system is designed to provide adequate control of the maximum excess reactivity expected during cycle operation. Because fuel reactivity is a maximum and control is a minimum at ambient temperature, the shutdown capability is evaluated assuming a cold, xenon-free core. The safety design basis requires that the core, in its maximum reactivity condition, be subcritical with the control rod of the highest worth fully withdrawn and all others fully inserted.

4.3.2.4.1 Shutdown Reactivity

To ensure that the safety design basis for shutdown is satisfied, an additional design margin is adopted: the reactor is subcritical with the control rod of highest worth fully withdrawn. A confirmation of this is provided in the supplemental reload licensing report for each cycle. A listing of these submittals is given in Table 4.3-1.

4.3.2.4.2 Reactivity Variations

The excess reactivity designed into the core is controlled by the control rod system supplemented by gadolinia-urania fuel rods. Gadolinia and enrichment distributions for these rods are given in Reference 3. Control rods are used during the cycle partly to compensate for burnup and partly to flatten the power distribution.

4.3.2.5 Control Rod Patterns and Reactivity Worths

For BWR plants, control rod patterns are not uniquely specified in advance; rather, during normal operation, the control rod patterns are selected from the measured core power distributions, within the constraints imposed by the rod worth minimizer described in Section 7.7.7. Control Requirements are discussed in Section 3.2.4 of Reference 1.

4.3.2.6 Shutdown of Reactor During Refueling

The core must be subcritical, with appropriate margin and allowing for the highest worth control rod withdrawn, at each step during core alterations. The shutdown margin is determined each cycle by using a safety-related BWR simulator code. See Section 15.1.4.4.1 for a description of the codes used and acceptance criteria.

4.3.2.7 Stability

4.3.2.7.1 Xenon Transients

BWRs do not have instability problems due to xenon. This has been demonstrated by operating BWRs for which xenon instabilities have never been

observed. Such instabilities would readily be detected by the local power range monitors, by special tests that have been conducted on operating BWRs in an attempt to force the reactor into xenon instability, and by calculations. All of these indicators have proved that xenon transients are highly damped in a BWR because of the large negative power coefficient.

Analyses and experiments conducted in this area are reported in Reference 2.

4.3.2.7.2 Thermal-Hydraulic Stability

This subject is covered in Section 4.4.

4.3.2.8 Vessel Irradiations

For discussion of RT_{NDT}, see Chapter 5.3.1.5.

4.3.3 ANALYTICAL METHODS

The analytical methods and nuclear data used to determine the nuclear
2012-020 characteristics are provided in Section 3 of Reference 1, Reference 5, Reference 6, and Reference 7. The qualification of these models is also noted in Reference 1, Reference 5, Reference 6, and Reference 7. Also see Chapter 15.0 for list of computer codes for DAEC specific analysis.

4.3.4 CHANGES

General Electric fuel design philosophy is based on three principles: standardization, evolution, and testing before use. This process has resulted in a series of 8 x 8 and 10 x 10 fuel designs. Details of these designs are provided in References 3, 4,
2012-020 5, 6, and 7.

REFERENCES FOR SECTION 4.3

1. General Electric Company, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest NRC approved version).

2. R. L. Crowther, Xenon Considerations in Design of Boiling Water Reactors, APED-5640, 1968.

3. General Electric Company, GE Fuel Bundle Designs, NEDE-31152P (latest NRC approved version).

4. General Electric Company, Licensing Criteria for Fuel Designs (Amendment 22 to NEDE-24011-P-A and Corresponding NRC Safety Evaluation), NEDO-31908, January 1991.

5. General Electric Company, GE12 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR-II), NEDE-32417P, December 1994.

6. General Electric Company, GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR-II), NEDC-32868P, December 1998.

2019-001 7. Global Nuclear Fuel, GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR-II), NEDC-33270P, Revision 7, October 2016.

Table 4.3-1

DUANE ARNOLD ENERGY CENTER SUPPLEMENTAL RELOAD LICENSING SUBMITTALS

Reload	Reference
1	General Electric Boiling Water Reactor Reload 1 Licensing Submittal, Duane Arnold Energy Center, NEDO-21082, November 1975. General Electric Boiling Water Reactor Reload 1 Licensing Submittal, Bypass Flow Holes Plugged, Duane Arnold Energy Center, NEDO-21082-01, January 1976.
2	General Electric Boiling Water Reactor Reload Number 2 Licensing Submittal, Duane Arnold Energy Center, NEDO-21082-02, 1977. General Electric Boiling Water Reactor Reload Number 2 Licensing Submittal Supplement 1, Partially Drilled Core, Duane Arnold Energy Center, NEDO-21082-02-1, 1977.
3	Boiling Water Reactor Reload 3 Licensing Amendment for Duane Arnold Atomic Energy Center, NEDO-24087, December 1977.
4	Supplemental Reload Licensing Submittal for Duane Arnold Atomic Energy Center Reload 4, NEDO-24234, January 1980.
5	Supplemental Reload Licensing Submittal for Duane Arnold Atomic Energy Center Reload 5, Y1003J01A18, January 1981.
6	Supplemental Reload Licensing Submittal for Duane Arnold Atomic Energy Center Reload 6, Y1003J01A46, March 1983.
7	Supplemental Reload Licensing Submittal for Duane Arnold Atomic Energy Center Reload 7, 23A1739, Revision 1, December 1984.
8	Supplemental Reload Licensing Submittal for Duane Arnold Atomic Energy Center Reload 8, 23A4812, Revision 0, September 1986.
9	Supplemental Reload Licensing Submittal for Duane Arnold Energy Center, Unit 1, Reload 9, Cycle 10, 23A5906, Revision 0, June 1988.
10	Supplemental Reload Licensing Submittal for Duane Arnold Energy Center, Reload 10, Cycle 11, 23A6450, Revision 1, August 1990.
11	Supplemental Reload Licensing Submittal for Duane Arnold Energy Center, Reload 11, Cycle 12, 23A7143, Rev. 0, February, 1992.
12	Supplemental Reload Licensing Submittal for Duane Arnold Energy Center, Reload 12, Cycle 13, 23A7210, Rev. 0, June, 1993.
13	Supplemental Reload Licensing Submittal for Duane Arnold Energy Center, Reload 13, Cycle 14, 24A5171, Rev. 1, August, 1995.
14	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 14, Cycle 15, 24A5369, Rev. 0, September 1996
15	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 15, Cycle 16, 24A5410, Rev. 0, March 1998
16	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 16, Cycle 17, J11-03517, SRLR, Rev. 0, October 1999.
17	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 17, Cycle 18, J11-03834SRLR1658, Rev. 0, April 2001

DUANE ARNOLD ENERGY CENTER SUPPLEMENTAL RELOAD LICENSING SUBMITTALS

	Reload	Reference
	18	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 18, Cycle 19, DRF 0000-0005-3066SRLR, Rev. 0, January 2003.
	19	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 19, Cycle 20, GNF 0000-0028-4296-SRLR, Rev. 0, February 2005.
	20	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 20, Cycle 21, GNF 0000-0051-8481-SRLR, Rev. 0, December, 2006.
	21	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 21, Cycle 22, GNF 0000-0082-2041-SRLR, Rev. 1, December 2008.
	22	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 22, Cycle 23, GNF 0000-0109-9269-SRLR, Rev. 0, August 2010.
2012-020	23	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 23, Cycle 24, GNF 0000-0136-4830-SRLR, Rev. 0, August 2012.
2015-004	24	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 24, Cycle 25, GNF 001N1159, Rev. 1, July 2014.
2017-001	25	Supplemental Reload Licensing Report for Duane Arnold Energy Center, Reload 25, Cycle 26, GNF 002N6817, Rev. 0, July 2016.
2019-001	26	Supplemental Reload Licensing Report for Duane Arnold, Reload 26, Cycle 27, GNF 004N2945, Rev. 0, June 2018.

4.4 THERMAL-HYDRAULIC DESIGN

4.4.1 DESIGN BASIS

4.4.1.1 Safety Design Bases

Thermal-hydraulic design of the core shall establish the following:

1. Actuation limits for the devices of the nuclear safety systems such that no fuel safety limit is calculated to be exceeded as a result of analyzed abnormal operational transient events.

2. The safety limits for use in evaluating the safety margin relating the consequences of fuel barrier failure to public safety.

3. That the nuclear system exhibits no inherent tendency toward divergent or limit cycle oscillations that would compromise the integrity of the fuel or nuclear system process barrier.

4.4.1.2. Power Generation Design Bases

The thermal-hydraulic design of the core shall provide the following operational characteristics:

1. The ability to achieve rated core power output throughout the design life of the fuel.

2. The flexibility to adjust core output over the range of plant load and load-maneuvering requirements in a stable, predictable manner without sustaining fuel damage.

4.4.1.3 Requirements for Steady-State Conditions

For purposes of maintaining adequate thermal margin during normal steady-state operation, the MCPR must not be less than the required MCPR operating limit, and the MLHGR must be maintained below the design LHGR for the plant. This does not specify the operating power nor does it specify peaking factors. These parameters are determined subject to a number of constraints, including the thermal limits given previously. The core and fuel design bases for steady-state operation (that is, MCPR and LHGR limits) have been defined to provide a margin between the steady-state operating conditions and any fuel damage condition to accommodate uncertainties and to ensure that no fuel damage results even during the worst anticipated transient condition at any time during plant life.

The design steady-state MCPR operating limit and APLHGR limit are given in the Core Operating Limits Report. Modifications for one recirculation loop operation are 2012-020 documented in References 10 and 14 and the current Supplemental Reload Licensing Report as shown in Table 4.3-1.

4.4.1.4 Requirements for Transient Conditions

The MCPR and MAPLHGR limits are established such that no safety limit is expected to be exceeded during the most severe moderate frequency Anticipated Operational Occurrences (AOO) event as defined in Reference 1.

4.4.1.5 Summary of Design Bases

The steady state operating limits have been established to assure that the design bases are satisfied for the most severe moderate frequency AOO. Demonstration that the steady state MCPR and MAPLHGR limits are not exceeded is sufficient to conclude that the design bases are satisfied .

4.4.2 DESCRIPTION OF THERMAL-HYDRAULIC DESIGN OF THE REACTOR CORE

4.4.2.1 Summary Comparison

An evaluation of plant performance from a thermal-hydraulic standpoint is provided in Section 4.4.3.

Cycle-specific parameters are provided in the supplemental reload licensing reports listed in Table 4.3-1.

4.4.2.2 Critical Power Ratio

A description of the critical power ratio and the model used to calculate this ratio is provided in Section 4.3.1 of Reference 1. Values of the fuel cladding integrity safety limit are given in Subsection 4.3.1.1 of Reference 1; operating MCPR limits are provided in the supplemental reload licensing reports listed in Table 4.3-1. Modifications of these 2012-020 limits for one recirculation pump operation is given in References 10 and 14. Operating MCPR limits for two recirculation loop and single loop operations are included in the Core Operating Limits Report.

4.4.2.3 MAPLHGR Limits

The models used to determine the MAPLHGR limits are described in Section 2.2 and S.2.2.3.2.1 of Reference 1.

4.4.2.4 Void Fraction Distribution

The void fraction is calculated each cycle. The value used in relatively slow transient analyses is provided in supplemental reload licensing reports listed in Table 4.3-1. This value is calculated within the transient code for rapid pressurization events and is not reported separately but is reflected in the plot of void reactivity in the figures given for these events in the supplemental reload licensing reports.

4.4.2.5 Core Coolant Flow Distribution

Hydraulic models and the core coolant flow distribution between the area inside the channel and the bypass region are given in Section 4.2 of Reference 1.

4.4.2.6 Core Pressure Drops and Hydraulic Loads

Models for core pressure drop are given in Section 4.2.4 of Reference 1.

4.4.2.7 Correlation and Physical Data

General Electric has obtained substantial amounts of physical data in support of the pressure drop and thermal-hydraulic loads. This information is given in Section 4.2.4 of Reference 1. Models used to calculate the heat-transfer coefficient are referenced in Section 4.2.5.3 of Reference 1.

4.4.2.8 Thermal Effects of Operational Transients

The evaluation of the core's capability to withstand the thermal effects resulting from anticipated operational transients is discussed in Chapter 15.

4.4.2.9 Uncertainties in Estimates

Uncertainties in thermal-hydraulic parameters considered in the statistical analysis that is performed to establish the fuel cladding integrity safety limits are documented in Reference 7.

4.4.2.10 Flux Tilt Considerations

The inherent design characteristics of the BWR are particularly well suited to handle perturbations due to flux tilt. The stabilizing nature of the moderator void coefficient effectively damps oscillations in the power distribution. In addition to this damping, the incore instrumentation system and the associated on-line computer provide the operator with prompt and readily available power distribution information. Thus, the operator can readily use control rods or other means to effectively limit the undesirable effects of flux tilting. Because of these features and capabilities, it is not necessary to allocate a specific peaking factor margin to account for flux tilt. If for some reason the

power distribution could not be maintained within normal limits using control rods, then the operating power limits would have to be reduced as described in the plant Core Operating Limits Report.

4.4.3 DESCRIPTION OF THERMAL-HYDRAULIC DESIGN OF THE REACTOR COOLANT SYSTEM

The thermal-hydraulic design of the reactor coolant system is described in this section.

4.4.3.1 Plant Configuration Data

The reactor coolant system is described in Chapter 5. Values from the reactor heat balance used in the safety analyses are given in Chapter 15.0.

4.4.3.2 Operating Restrictions on Pumps

Section 4.4.3.3 gives the operating limits imposed on the recirculation pumps by cavitation, pump loads, bearing design flow starvation, and pump speed.

4.4.3.3 Power-Flow Operating Map

A BWR must operate with certain restrictions because of pump net positive suction head (NPSH), overall plant control characteristics, core thermal power limits, etc. The latest power flow map is included in the Core Operating Limits Report. The operating envelope includes the region bounded by the MELLLA boundary, the rated power line, and the 105% of rated core flow line. Reference 12 provides the analytical bases for operation of the DAEC under this operating envelope, which is validated as part of the cycle-dependent reload analysis. The average power range monitor, rod block monitor, and Technical Specification improvement (ARTS) program provided the system and instrumentation improvements and operating procedure changes which permit steady-state operation in the region of the power flow map at off-rated conditions of power and/or flow (References 4 and 12). The nuclear system equipment, nuclear instrumentation, and the reactor protection system, in conjunction with operating procedures, maintain operations within the area of this map for normal operating conditions. The boundaries of this map are as follows:

1. Natural Circulation Line

   The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.

2. Minimum Pump Speed Line

   Startup operations of the plant are normally carried out with the recirculation pumps operating at approximately 20% speed. The operating state for the reactor follows this line for the normal control rod withdrawal sequence.

3. Operational Upper Loadline Limit

   This is a power-flow line that is bounded by the MELLLA boundary up to 100% rated power/99% rated flow.

4. Maximum Thermal Power Line

   The maximum allowable thermal power line is 100% rated thermal power between 99% and 105% rated core flow.

5. Low Feedwater Protection Line

   This line results from the recirculation pump and jet pump NPSH requirements. When feedwater flow drops below 20% of rated flow, the recirculation pumps are automatically tripped to 20% speed.

6. Exclusion Zone

   The exclusion zone is defined in Reference 6 and updated in the supplemental reload submittals listed in Table 4.3-1. The DAEC shall not have a planned entry into this area of the power/flow map due to the reduced margin to thermal hydraulic instability.

7. Buffer Zone

   The buffer zone is defined in the Core Operating Limits Report. Planned entry into the buffer zone is allowed if acceptable results are demonstrated from the stability monitor in use at the DAEC.

4.4.3.4 Temperature-Power Operating Map

Not applicable to the DAEC (applies to pressurized-water reactors).

4.4.3.5 Thermal-Hydraulic Characteristics Summary Table

The thermal-hydraulic characteristics used in the safety analyses are given in Table 2-1 of Reference 7 and in the supplemental reload submittals listed in Table 4.3-1. See also Chapter 15.0.

4.4.4 EVALUATION

The design basis employed for the thermal-hydraulic characteristics incorporated in the core design, in conjunction with the plant equipment characteristics, nuclear instrumentation, and the reactor protection system, is to require that no fuel damage occurs during normal operation or during abnormal operational transients. Analyses have demonstrated that the applicable thermal-hydraulic limits are not exceeded.

4.4.4.1 Critical Power

The GEXL critical power correlation used in thermal-hydraulic evaluations is discussed in Section 4.3.1 of Reference 1.

4.4.4.2 Core Hydraulics

Core hydraulic models and correlations are discussed in Section 4.3.2 of Reference 1.

4.4.4.3 Influence of Power Distributions

The influence of power distributions on the thermal-hydraulic design is discussed in Section 4.3.3 of Reference 1.

4.4.4.4 Core Thermal Response

The thermal response of the core for accidents and expected transient conditions is discussed in Chapter 15.

4.4.4.5 Analytical Methods

The analytical methods, thermodynamic data, and hydrodynamic data used in determining the thermal-hydraulic characteristics of the core are documented in Chapter 15.0.

4.4.4.6 Thermal-Hydraulic Stability Analysis

The original Option I-D DAEC thermal-hydraulic stability analysis is given in Reference 6. This analysis was then updated with the methodology described in
2019-001 Reference 11, and again with the methodology described in Reference 13 and Reference 15. The cycle-specific results are found in the supplemental reload submittals listed in Table 4.3-1 and the Core Operating Limits Report. See also Chapter 15.3.4.

4.4.5 TESTING AND VERIFICATION

The testing and verification techniques used to ensure that the planned thermal-hydraulic design characteristics of the core will remain within required limits throughout the life of the core are discussed in Chapter 14 and in the Technical Specifications.

4.4.6 INSTRUMENTATION REQUIREMENTS

The reactor vessel instrumentation monitors the key reactor vessel operating parameters during planned operations. This ensures sufficient control of the parameters. The reactor vessel sensors are discussed in Section 7.6.4.

REFERENCES FOR SECTION 4.4

1. General Electric Company, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest NRC approved version).

2. Deleted.

3. Deleted.

4. General Electric Company, General Electric BWR Licensing Report: Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for the Duane Arnold Energy Center, NEDO-30813, March 1985.

5. Deleted.

6. Application of the "Regional Exclusion with Flow-Biased APRM Neutron Flux Scram", Stability Solution (Option I-D) to the Duane Arnold Energy Center, GENE-A00-04021-01, September, 1995.

7. General Electric Company, GE Fuel Bundle Designs, NEDE-31152P (latest approved version).

8. General Electric Company, Duane Arnold Energy Center GE12 Fuel Upgrade Project, NEDC-32915P, Revision 0, November, 1999.

9. Deleted.

10. General Electric Report, Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate, NEDC-32980P, Nov. 2000.

11. General Electric Report, ODYSY Application for Stability Licensing Calculations, NEDC 32992P, October 2000.

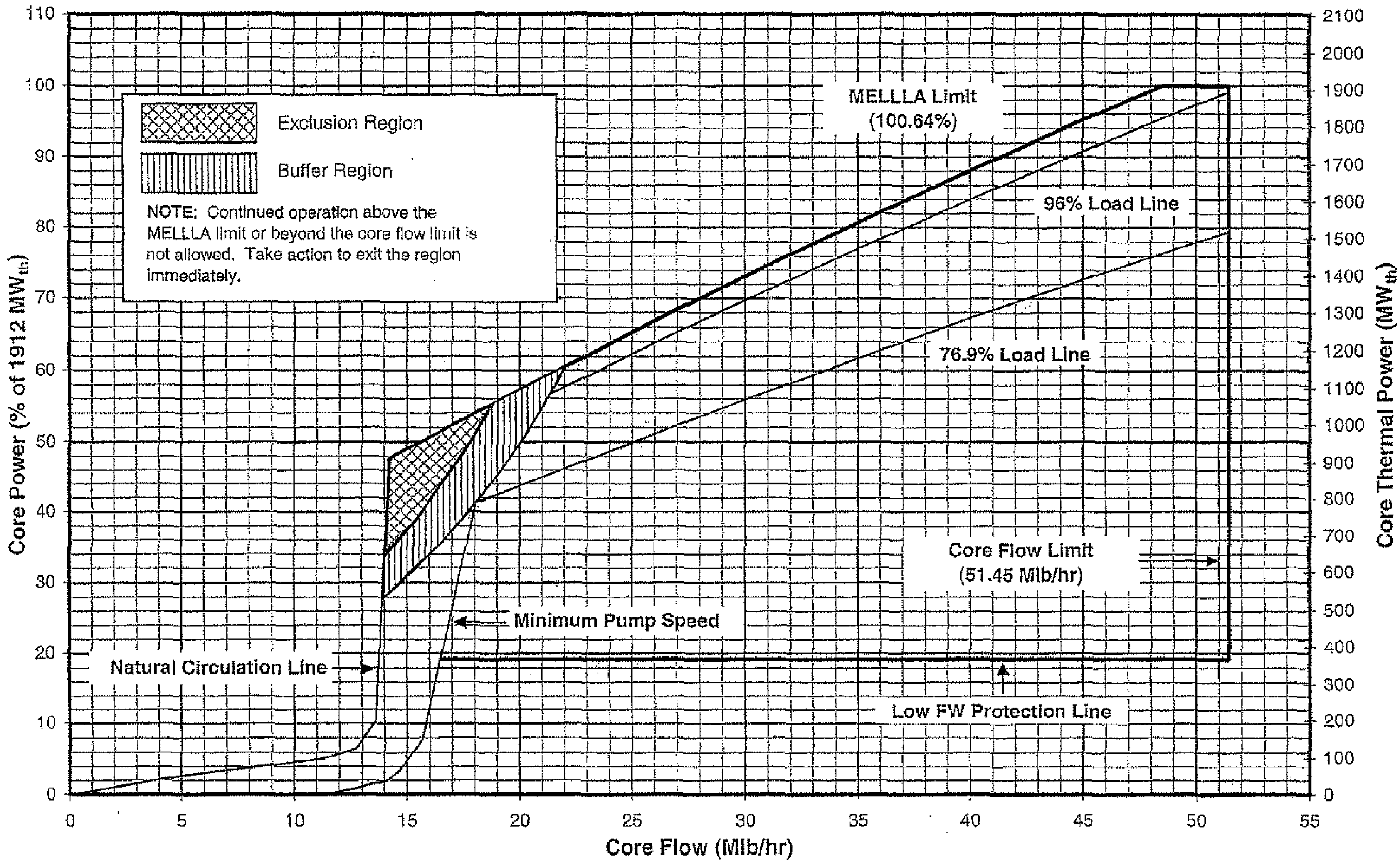
12. GE Hitachi Nuclear Energy, Safety Analysis Report for Duane Arnold Energy Center Increased Core Flow, NEDC-33439P, Revision 3, August, 2009.

13. GE Hitachi Nuclear Energy, ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions, NEDE-33213P-A, April 2009.

2012-020

14. Global Nuclear Fuel, GNF2 Fuel Design Cycle-Independent Analyses for Duane Arnold Energy Center, NEDC-33704P, Revision 0, August, 2012.

2019-001 15. GE Hitachi Nuclear Energy, GEH Simplified Stability Solution (GS3), NEDE-33766P-A, Revision 1, March 2015.



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Power Flow Map – Core Thermal Power Versus Core Flow

Figure 4.4-1

4.5 REACTOR MATERIALS

4.5.1 CRD SYSTEM STRUCTURAL MATERIALS

Factors that determine the choice of CRD system construction materials are discussed below.

The index tube must withstand the locking and unlocking action of the collet fingers. A compatible bearing combination must be provided that is able to withstand moderate misalignment forces. The reactor environment limits the choice of materials suitable for corrosion resistance. The column and tensile loads can be satisfied by an annealed 300 series stainless steel. The wear and bearing requirements are satisfied by Malcomizing the completed tube to obtain suitable corrosion resistance. Because misalignment tends to cause chafing in the semispherical contact area, the entire part is protected by a thin vapor-deposited chromium plating (electrolized). This plating also prevents the galling of the threads attaching the coupling spud to the index tube.

Inconel-750 is used for the collet fingers, which must function as leaf springs when cammed open to the unlocked position. Colmonoy-6 hard facing provides a long-wearing surface, adequate for design life, to the area contacting the index tube and unlocking cam surface of the guide cap.

Graphitar-14 is selected for seals and bushings on the drive piston and stop piston. The material is inert and has a low friction coefficient when water lubricated.

Because a loss of seal strength is experienced at higher temperatures, the drive is supplied with cooling water to hold temperatures below 250▪F. The Graphitar is relatively soft, which is advantageous when an occasional particle of foreign matter reaches a seal. The resulting scratches in the seal reduce sealing efficiency until worked smooth, but the drive design can tolerate considerable water leakage past the seals into the reactor vessel.

All drive components exposed to reactor vessel water are made of AISI 300 series stainless steel except the following:

1. Seals and bushings on the drive piston and stop piston are Graphitar-14.

2. All springs and members requiring spring action (collet fingers, coupling spud, and spring washers) are made of Inconel-750.

3. The ball check valve is a Haynes Stellite cobalt-based alloy.

4. Elastomeric O-ring seals are ethylene propylene.

5. Collet piston rings are Haynes-25 alloy.

6. Certain wear surfaces are hard faced with Colmonoy-6.

7. Nitriding by a proprietary Malcomizing process, electrolyzing (a vapor deposition of chromium), and chromium plating are used in certain areas where resistance to abrasion is necessary.

8. The drive piston head is made of Armco 17-4Ph.

Pressure-containing portions of the drives are designed and fabricated in accordance with requirements of Section III of the ASME Code.

4.5.2 REACTOR INTERNAL MATERIALS

The reactor internal materials are described in Section 3.9.5. Fuel assembly materials are discussed in Section 4.2.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

The reactivity control systems consist of control rods, the CRD system, supplementary reactivity control for the initial core, the standby liquid control system, and the recirculation flow control system. The recirculation flow control system is described in Section 7.7.5. The standby liquid control system is described in Section 9.3.4.

4.6.1 CRD SYSTEM DESIGN AND DESCRIPTION

4.6.1.1 <u>CRD System Design</u>

4.6.1.1.1 Safety Design Bases

The CRD mechanical system meets the following safety design bases:

1. The design provides for a sufficiently rapid control rod insertion so that no fuel damage results from any abnormal operating transient.

2. The design includes positioning devices, each of which individually supports and positions a control rod.

3. Each positioning device

   a. Prevents its control rod from withdrawing as a result of a single malfunction.

   b. Is individually operated so that a failure in one positioning device does not affect the operation of any other positioning device.

   c. Is individually energized when rapid control rod insertion (scram) is signaled, so that a failure of power sources external to the positioning device does not prevent other positioning devices' control rods from being inserted.

4. Is designed so that no single failure of a component will prevent its control rod from being inserted.

4.6.1.1.2 Power Generation Design Basis

The CRD design provides for positioning the control rods to control power generation in the core.

4.6.1.2 CRD System Description

The CRD system controls gross changes in core reactivity by incrementally positioning neutron-absorbing control rods within the reactor core, in response to manual control signals. It is also designed to quickly shut down the reactor (scram) in emergency situations by rapidly inserting withdrawn control rods into the core in response to a manual or automatic signal. The CRD system consists of locking piston CRD mechanisms and the CRD hydraulic system (including power supply and regulation, hydraulic control units, interconnecting piping, instrumentation, and electrical controls).

4.6.1.2.1 CRD Mechanisms

The CRD mechanism (drive) used for positioning the control rod in the reactor core is a double-acting, mechanically latched, hydraulic cylinder using demineralized water as its operating fluid. The individual drives are mounted on the bottom head of the reactor pressure vessel. See Section 3.9.4.1.1.

The drives are capable of inserting or withdrawing a control rod at a slow, controlled rate, in addition to providing rapid insertion when required. A mechanism on the drive locks the control rod at 6-in. increments of stroke over the length of the core.

4.6.1.2.2 Drive Components and Materials of Construction

Drive Components

The operating principle of a drive is shown in Figure 3.9-2. Figures 3.9-3 and 3.9-4 illustrate the drive in more detail. The main components of the drive and their functions are described in Section 3.9.4.1.1.

4.6.1.2.3 CRD Hydraulic System

The CRD hydraulic system (Figure 3.9-5) supplies and controls the pressure and flow to and from the drives. One supply subsystem supplies water to the hydraulic control units at the correct flow. Each hydraulic control unit controls the flow to and from a drive. The water discharged from the drives during a scram flows through the hydraulic control units to the scram discharge volume. See Section 3.9.4.1.2.

4.6.1.2.3.1 CRD System Operation

The CRD system performs rod insertion, rod withdrawal and scram. These operational functions are described in Section 3.9.4.1.3.

The requirements for the CRD system have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. The CRD system is intended to provide sufficient control of core reactivity so that the core could be made subcritical with the strongest rod fully withdrawn.

The Technical Specifications require that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and then disarmed electrically*, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a non-fully-inserted position, that position must be consistent with the shutdown reactivity limitation stated in the Technical Specifications. This ensures that the core can be shut down at all times with the remaining control rods, assuming that the strongest operable control rod does not insert. Inoperable bypassed rods not in compliance with BPWS are separated by 2 or more operable control rods in any direction, including diagonal. The use of jumpers in the rod position indication system to substitute for a failed full-in or full-out position switch will not be limited as long as the actual position is known.

Control rod drop accidents as discussed in Chapter 15 can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. The absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper functioning of the Rod Worth Minimizer.

The control rod housing support restricts the outward movement of a control rod to less than 3 in. in the extremely remote event of a housing failure. The amount of reactivity that could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. This support is not required if the reactor coolant system is at atmospheric pressure, because there would then be no driving force to rapidly eject a drive housing. In addition, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, because the reactor would remain subcritical even, in the event of complete ejection of the strongest control rod.

4.6.1.2.3.2 Scram Reliability

High scram reliability is the result of a number of features of the CRD system. The following are examples:

1. Two sources of scram energy are used to insert each control rod when the reactor is operating--accumulator pressure and reactor vessel pressure.

2. Each drive mechanism has its own scram and pilot valves so only one drive can be affected if a scram valve fails to open. A pilot valve is provided for each drive. Both solenoid coils must be deenergized to allow a vent path to initiate a scram.

3. The reactor protection system and the hydraulic control units are designed so that the scram signal and mode of operation override all others.

4. The collet assembly and index tube are designed so that they will not restrain or prevent control rod insertion during scram.

5. The scram discharge volume is monitored for accumulated water and will scram the reactor before the volume is reduced to a point that could interfere with a scram.

4.6.1.2.3.3 Control Rod Support and Operation

As described above, each control rod is independently supported and controlled as required by the safety design bases.

4.6.1.2.4 CRD Housing Supports

See Section 3.9.4.1.4. CRD housing failure is further discussed in Section 4.6.2.2.

4.6.1.2.5 Control Rods

The control rods perform the dual function of power shaping and reactivity control. Power distribution in the core is controlled during the operation of the reactor by manipulating selected patterns of control rods.

The DAEC core contains 89 control rod assemblies. Current control rod inventory
2015-004 consists of GE Duralife D160 control rods, GE Duralife D230 control rods, GE Marathon control
2012-020 rods and Westinghouse CR-99 control rods. D160 control rods were first installed for Cycle 9 (DDC 1145). D230 control rods were initially installed for Cycle 11 (DCP 1490). GE Marathon
2012-020 control rods were initially installed for Cycle 20 (ECP 1702). Westinghouse CR-99 control rods were initially installed for Cycle 24 (EC 271695).

The original GE control rods consisted of a central cruciform commercial-grade 304 stainless steel tie rod with four sheathed wings, each containing 21 absorber tubes filled with boron carbide (B_4C) powder. The boron carbide power in the absorber tubes is compacted to about 70% of its theoretical density. This design used stellite material containing cobalt for pins and rollers. The U-shaped sheaths are resistance welded to the center post, handle, and castings to form a rigid housing to contain the absorber rods filled with boron carbide. The U-shaped sheaths are perforated to allow the coolant to circulate freely about the absorber tubes. Evaluation and approval of the original plant equipment was completed during plant construction
2012-020 and licensing. A typical GE control blade is shown in Figure 4.6-2.

The D160 design replaced the three outer absorber tubes in each wing with hafnium absorber rods. A high purity form of 304 stainless steel was used to reduce control rod cracking. Stellite materials were replaced to reduce cobalt activation products. Description, evaluation and approval of the D160 design is contained in References 7 and 8.

The D230 design reduced the total number of boron carbide absorber tubes to 60 per control rod and employed hafnium strips and plates to provide the remaining neutron absorber material. Extended lifetime in the D230 is achieved by replacing the top 6 inches of the absorber tubes with a hafnium plate and increasing the overall thickness of the absorber tubes (to add more boron carbide powder). Description, evaluation, and approval of the D230 design is contained in References 9 and 10.

The GE Marathon design eliminated the sheathing and is composed of a stainless steel segmented tie rod to which "square" fourteen absorber tubes are full-length welded to form the four blades. The absorber rod closest to the tie rod is empty of absorber material (filled with helium). The next ten absorber rods contain encapsulated boron carbide power. The outer three absorber tubes again contain hafnium absorber rods. Description, evaluation, and approval of the Marathon design is contained in Reference 11 and 12.

2012-020 The Westinghouse CR-99 design consists of four stainless steel sheets welded together to from a cruciform shaped rod. Each sheet has horizontally drilled holes to contain the absorber material (HIP B_4C pins). The HIP pins have the theoretical density of B_4C. Between the pins and the stainless steel wall a gap is designed to avoid hard contact between the swelling absorber material and the stainless steel wall under operation to the nuclear end of life. Description, evaluation and approval of the CR-99 design is contained in Reference 14. A typical Westinghouse control blade is shown in Figure 4.6-7.

All designs are dimensionally compatible and have undepleted reactivity worth within 5% of the original design control rods. Thus these different designs are considered to be equivalent in form, fit and function and are fully interchangeable.

The control rods are 9.84 in. maximum total span and are separated uniformly throughout the core on a 12-in. pitch. Each control rod is surrounded by four fuel assemblies (see Section 4.2).

The main structural member of a control rod is made of Type 304 stainless steel and consists of a top handle, a bottom casting with a velocity limiter and CRD coupling, and a vertical cruciform center post. The top handle, bottom casting, and center post are welded into a single skeletal structure. Rollers at the top and bottom of the control rod guide the control rod as it is inserted and withdrawn from the core. The control rods are cooled by the core bypass flow. Operating experience has shown that control rods constructed as described above are not susceptible to dimensional distortions.

2012-020 References 13 and 14 contain discussion of various mechanical and nuclear design considerations that define the life-limiting mechanism for control blades.

4.6.1.2.5.1 Control Rod Velocity Limiter

The control rod velocity limiter (see Figure 4.6-6) is an integral part of the bottom assembly of each control rod. This engineered safeguard protects against a high reactivity insertion rate by limiting the control rod velocity in the event of a control rod drop accident. It is a one-way device in that the control rod scram velocity is not significantly affected, but the control rod dropout velocity is reduced to a permissible limit.

The velocity limiter is in the form of two nearly mated conical elements that act as a large clearance piston inside the control rod guide tube. The lower conical element is separated from the upper conical element by four radial spacers 90 degrees apart and is at a 15-degree angle relative to the upper conical element, with the peripheral separation less than the central separation.

The hydraulic drag forces on a control rod are proportional to approximately the square of rod velocity and are negligible at normal rod withdrawal or rod insertion speeds. However, during the scram stroke, the rod reaches high velocity, and the drag forces must be overcome by the drive mechanism.

To limit control rod velocity during dropout but not during scram, the velocity limiter is provided with a streamlined profile in the scram (upward) direction. Thus, when the control rod is scrammed, water flows over the smooth surface of the upper conical element into the annulus between the guide tube and the limiter. In the dropout direction, however, water is trapped by the lower conical element and discharged through the annulus between the two conical sections. Because this water is jetted in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created, thereby slowing the descent of the control rod assembly to less than 5 fps at 70°F.[2]

4.6.1.2.6 Safety Evaluation

4.6.1.2.6.1 Evaluation Of Control Rods

The description shows that the control rods meet the design bases. The description also shows how the control-rod-to-drive-coupling unit meets the design bases.

4.6.1.2.6.2 Evaluation of Control Rod Velocity Limiter

The control rod velocity limiter limits the free-fall velocity of the control rod to a value that cannot result in nuclear system process barrier damage. This velocity is evaluated by the rod drop accident analysis in Chapter 15.2.

4.6.2 EVALUATION OF THE CRD SYSTEM

4.6.2.1 Evaluation of Scram Time

The scram function may be divided into two parts: initiation and function. All scram-related instrumentation meets the intent of IEEE 279.

The reactor protection system initiation trip systems perform the initiation function. Functional diversity inherent in the reactor protection system provides a high order of resistance to common-mode failures. The actuation function is performed by the CRD system and actuation logic, which are highly redundant. Great care has been exercised in their design and application, and they are tested thoroughly and frequently. The actual design has been carefully refined as experience shows the need for modifications. Thus, the design is very mature and thoroughly standardized. As a result of close scrutiny over many years by many designers and analysts, the opportunity for undetected common-mode failure appears to be remote, and failure due to coincident random events has a very low probability.

The design and performance of the BWR scram system, including both the initiation and actuation functions, exhibit high integrity and reliability of the total protection function such that common-mode failures that could negate scram action are not credible.

The current design of the BWR scram system is satisfactory with due consideration of its reliability, availability, and performance. However, a design change to incorporate the recirculation pump trip was made as it was felt that an overall increase in plant safety was affected. The recirculation pumps will be tripped from a high reactor pressure event. This will provide an effective shutdown of the plant as described in NEDO-10349[3] and discussed in Section 7.2.1.2.3.

The rod scram function of the CRD system provides the negative reactivity insertion required by the safety design basis.

4.6.2.2 Analysis of Malfunctions Relating to Rod Withdrawal

There are no known single malfunctions that cause the unplanned withdrawal of even a single control rod. However, if multiple malfunctions are postulated, the results show that an unplanned rod withdrawal can occur at withdrawal speeds that vary with the combination of malfunctions postulated. In all cases, the subsequent withdrawal speeds are less than that assumed in the rod drop accident analysis as discussed in Chapter 15. Therefore, the physical and radiological consequences of such rod withdrawals are less than those analyzed in the rod drop accident.

Drive Housing Fails at Attachment Weld

The bottom head of the reactor vessel has a penetration with an internal nozzle for each CRD location. A drive housing is raised into position inside each penetration and fastened to the top of the internal nozzle with a J-weld. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing. The housing material is seamless, Type 304 stainless steel pipe with a minimum tensile strength of 75,000 psi. The basic failure considered here is a complete circumferential crack through the housing wall at an elevation just below the J-weld.

Static loads on the housing wall include the weight of the drive and the control rod, the weight of the housing below the J-weld, and the reactor pressure acting on the 6-in. diameter cross-sectional area of the housing and the drive. Dynamic loading results from the reaction force during drive operation.

If the housing were to fail as described, the following sequence of events is foreseen. The housing would separate from the vessel. The control rod, drive, and housing would be blown downward against the support structure by reactor pressure acting on the cross-sectional area of the housing and the drive. The downward motion of the drive and associated parts would be determined by the gap between the bottom of the drive and the support structure and by the deflection of the support structure under load. In the current design, maximum deflection is approximately 3 in. If the collet were to remain latched, no further control rod ejection would occur;[4] the housing would not drop far enough to clear the vessel penetration. Reactor water would leak at a rate of approximately 440 gpm through the 0.06-in. diametral clearance between the housing and the vessel penetration.

If the basic housing failure were to occur while the control rod is being withdrawn (this is a small fraction of the total drive operating time) and if the collet were to stay unlatched, the following sequence of events is foreseen. The housing would separate from the vessel. The drive and housing would be blown downward against the CRD housing support.

Calculations indicate that the steady-state rod withdrawal velocity would be 0.3 fps. During withdrawal, pressure under the collet piston would be approximately 250 psi greater than the pressure over it. Therefore, the collet would be held in the unlatched position until driving pressure was removed from the pressure-over port.

Rupture of Hydraulic Line(s) to Drive Housing Flange

There are three types of possible rupture of hydraulic lines to the drive housing flange: (1) pressure-under line break, (2) pressure-over line break, and (3) coincident breakage of both of these lines.

For the case of a pressure-under line break, a partial or complete circumferential opening is postulated at or near the point where the line enters the housing flange. Failure is more likely to occur after another basic failure wherein the drive housing or housing flange separates from the reactor vessel. A failure of the housing, however, does not necessarily lead directly to a failure of the hydraulic lines.

If the pressure-under line were to fail and if the collet were latched, no control rod withdrawal would occur. There would be no pressure differential across the collet piston and, therefore, no tendency to unlatch the collet. Consequently, the associated control rod could not be inserted or withdrawn.

The ball check valve is designed to seal off a broken pressure-under line by using reactor pressure to shift the check ball to its upper seat. If the ball check valve were prevented from seating, reactor water would leak to the atmosphere. Because of the broken line, cooling water could not be supplied to the drive involved. A loss of cooling water would cause no immediate damage to the drive. However, prolonged exposure of the drive to temperatures at or near reactor temperature could lead to the deterioration of material in the seals. High temperature would be indicated to the operator by the thermocouple in the position indicator probe. A second indication would be high cooling-water flow.

If the basic line failure were to occur while the control rod is being withdrawn and if the collet were to remain open, calculations indicate that the steady-state control rod withdrawal velocity would be 2 fps. However, the hydraulic force would not be sufficient to hold the collet open, and spring force normally would cause the collet to latch and stop rod withdrawal.

For the case of a pressure-over line break, a complete break of the line is postulated at or near the point where it enters the housing flange. If the line were to break, pressure over the drive piston would drop from reactor pressure to atmospheric pressure. Any significant reactor pressure (approximately 500 psig or greater) would act on the bottom of the drive piston and fully insert the drive. Insertion would occur regardless of the operational mode at the time of the failure. After full insertion, reactor water would leak past the stop piston seals, the contracting seals on the drive piston, and the collet piston seals. The leakage would exhaust to the atmosphere through the broken pressure-over line. The leakage rate at 1000 psi reactor pressure is estimated to be 1 to 3 gpm; however, with the seals of the stop piston removed, the leakage rate could be as high as 10 gpm, based on experimental measurements.[5] If the reactor were hot, drive temperature would increase. This situation would be indicated to the reactor operator by the drift alarm and light, by the fully inserted drive, by a high drive temperature (alarmed and printed out on a recorder in the control room), and by the operation of the drywell sump pump.

For the simultaneous breakage of the pressure-over and pressure-under lines, pressures above and below the drive piston would drop to zero, and the ball check valve would close the broken pressure-under line. Reactor water would flow from the annulus outside the drive, through the vessel ports, and to the space below-the drive piston. As in the case of pressure-over line breakage, the drive would then insert at a speed dependent on reactor pressure. Full insertion would occur regardless of the operational mode at the time of failure. Reactor water would leak past the drive seals and out the broken pressure-over line to the atmosphere, as described above. Drive temperature would increase. Indication in the control room would include the drift alarm and light, the fully inserted drive, the high drive temperature (alarmed and printed out on a recorder in the control room), and the operation of the drywell sump pump.

All Drive Flange Bolts Fail in Tension

Each control rod drive is bolted to a flange at the bottom of a drive housing. The flange is welded to the reactor vessel. Bolts are made of steel, with a minimum tensile strength of 125,000 psi. Each bolt has a minimum allowable load capacity of 15,200 lb; the capacity of the eight bolts is at least 121,600 lb. As a result of the reactor design pressure of 1250 psig, the major load on all eight bolts is 30,400 lb.

If a progressive or simultaneous failure of all bolts were to occur, the drive would separate from the housing. The control rod and the drive would be blown downward against the support structure. Impact velocity and support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act on the drive cross-sectional area only and the housing would remain attached to the reactor vessel. The drive would be isolated from the cooling-water supply. Reactor water would flow downward past the velocity limiter piston, through the large drive filter, and into the annular space between the thermal sleeve and the drive. For worst-case leakage calculations, the large filter is assumed to be deformed or swept out of the way so that it would offer no significant flow restriction. At a point near the top of the annulus, where pressure would have dropped to 350 psi, the water would flash to steam and cause choke-flow conditions. Steam would flow down the annulus and out the space between the housing and the drive flanges to the atmosphere. Steam formation would limit the leakage rate to approximately 840 gpm.

If the collet were latched, control rod ejection would be limited to the distance the drive can drop before coming to rest on the support structure. There would be no tendency for the collet to unlatch, because pressure below the collet piston would drop to zero. Pressure forces, in fact, exert 1435 lb to hold the collet in the latched position.

If the bolts failed during control rod withdrawal, pressure below the collet piston would drop to zero. The collet, with a 1650-lb return force, would latch and stop rod withdrawal.

Weld Joining Flange to Housing Fails in Tension

The failure considered is a crack in or near the weld that joins the flange to the housing. This weld extends through the wall and completely around the housing. The flange material is forged, Type 304 stainless steel, with a minimum tensile strength of 75,000 psi. The housing material is seamless, Type 304 stainless steel pipe, with a minimum tensile strength of 75,000 psi. The conventional, full-penetration weld of Type 308 stainless steel has a minimum tensile strength approximately the same as that for the parent metal. The design pressure and temperature are 1250 psig and 575°F. Reactor pressure acting on the cross-sectional area of the drive; the weight of the control rod, drive, and flange; and the dynamic reaction force during drive operation results in a maximum tensile stress at the weld of approximately 6000 psi.

If the basic flange-to-housing joint failure occurred, the flange and the attached drive would be blown downward against the support structure. The support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act only on the drive cross-sectional area. A lack of differential pressure across the collet piston would cause the collet to remain latched and limit control rod motion to approximately 3 in. Downward drive movement would be small; therefore, most of the drive would remain inside the housing. The pressure-under and pressure-over lines are flexible enough to withstand the small displacement and remain attached to the flange. Reactor water would follow the same leakage path described above for the flange-bolt failure, except that exit to the atmosphere would be through the gap between the lower end of the housing and the top of the housing flange. Water would flash to steam in the annulus surrounding the drive. The leakage rate would be approximately 840 gpm.

If the basic failure were to occur during control rod withdrawal (a small fraction of the total operating time) and if the collet were held unlatched, the flange would separate from the housing. The drive and flange would be blown downward against the support structure. The calculated steady-state rod withdrawal velocity would be 0.13 fps. Because pressure-under and pressure-over lines remain intact, driving water pressure would continue to the drive, and the normal exhaust line restriction would exist. The pressure below the velocity limiter piston would drop below normal as a result of leakage from the gap between the housing and the flange. This differential pressure across the velocity limiter piston would result in a net downward force of approximately 70 lb. Leakage out of the housing would greatly reduce the pressure in the annulus surrounding the drive. Thus, the net downward force on the drive piston would be less than normal. The overall effect of these events would be to reduce rod withdrawal to approximately one-half of normal speed. With a 560-psi differential across the collet piston, the collet would remain unlatched; however, it should relatch as soon as the drive signal is removed.

Housing Wall Ruptures

This failure is a vertical split in the drive housing wall just below the bottom head of the reactor vessel. The flow area of the hole is considered equivalent to the annular area between the drive and the thermal sleeve. Thus, flow through this annular area, rather than flow through the hole in the housing, would govern leakage flow. The housing is made of Type 304 stainless steel seamless pipe, with a minimum tensile strength of 75,000 psi. The maximum hoop stress of 11,900 psi results primarily from the reactor design pressure (1250 psig) acting on the inside of the housing.

If such a rupture were to occur, reactor water would flash to steam and leak through the hole in the housing to the atmosphere at approximately 1030 gpm. Choke-flow conditions would exist, as described above for the flange-bolt failure. However, leakage flow would be greater because flow resistance would be less; that is, the leaking water and steam would not have to flow down the length of the housing to reach the atmosphere. A critical pressure of 350 psi causes the water to flash to steam.

No pressure differential across the collet piston would tend to unlatch the collet, but the drive would insert as a result of a loss of pressure in the drive housing causing a pressure drop in the space above the drive piston.

If this failure occurred during control rod withdrawal, drive withdrawal would stop, but the collet would remain unlatched. The drive would be stopped by a reduction of the net downward force acting on the drive line. The net force reduction would occur when the leakage flow of 1030 gpm reduces the pressure in the annulus outside the drive to approximately 540 psig, thereby reducing the pressure acting on top of the drive piston to the same value. A pressure differential of approximately 710 psi would exist across the collet piston and hold the collet unlatched as long as the operator held the withdraw signal.

Flange Plug Blows Out

To connect the vessel ports with the bottom of the ball check valve, a hole of 0.75 in. in diameter is drilled in the drive flange. The outer end of this hole is sealed with a plug of 0.812 in. in diameter and 0.250 in. thickness. A full-penetration, Type 308 stainless steel weld holds the plug in place. The postulated failure is a full circumferential crack in this weld and subsequent blowout of the plug.

If the weld were to fail, the plug were to blow out, and the collet remained latched, there would be no control rod motion. There would be no pressure differential across the collet piston acting to unlatch the collet. Reactor water would leak past the velocity limiter piston, down the annulus between the drive and the thermal sleeve, through the vessel ports and drilled passage, and out the open plug hole to the atmosphere at approximately 320 gpm. Leakage calculations assume only liquid flows from the flange. Actually, hot reactor water would flash to steam, and choke-flow conditions would exist. Thus, the expected leakage rate would be lower than the calculated value. Drive temperature would increase and initiate an alarm in the control room.

If this failure were to occur during control rod withdrawal and if the collet were to stay unlatched, calculations indicate that control rod withdrawal speed would be approximately 0.24 fps. Leakage from the open plug hole in the flange would cause reactor water to flow downward past the velocity limiter piston. A small differential pressure across the piston would result in an insignificant driving force of approximately 10 lb, tending to increase withdraw velocity.

A pressure differential of 295 psi across the collet piston would hold the collet unlatched as long as the driving signal was maintained.

Flow resistance of the exhaust path from the drive would be normal because the ball check valve would be seated at the lower end of its travel by pressure under the drive piston.

Pressure Regulator and Bypass Valves Fail Closed (Reactor Pressure, 0 psig)

By regulating the amount of water from the supply pump that is bypassed back to the reactor, pressure in the drive water header, which is used to supply all drives, is controlled. This control is achieved primarily with the drive water pressure control valves and secondarily with

the pressure stabilizing valves. Two drive water control valves are arranged in parallel. One is motor operated and can be adjusted from the control room. It is normally in service and is partially open to maintain pressure in the header equal to the reactor pressure plus 260 psig. The other valve is hand operated and normally closed. It can be operated locally when the motor-operated valve is out of service.

The pressure stabilizing valves are solenoid operated and have built-in needle valves to adjust flow. The two valves are arranged in parallel between the drive water header and the return line to the reactor. One valve is set to bypass 2 gpm, and it closes when any drive is given a withdraw signal and diverts flow to the drive being operated rather than back to the reactor. This serves to maintain a relatively constant header pressure. Similarly, the other valve is set to bypass 4 gpm and closes when an insert signal is given to any drive.

This failure occurs when all these valves are closed and a maximum supply pump head of 1510 psi is in the drive water header with minimum recirculation flow. Most of the bypass flow normally passes through the motor-operated valve; therefore, the closure of this valve is critical.

Because exhaust line pressure is lowest when reactor pressure is zero, this reactor condition is also assumed. If the valve closure failure described above were to occur during control rod withdrawal, calculations indicate that steady-state withdrawal speed would be approximately 0.5 fps or twice normal velocity. A pressure differential of 1670 psi would exist across the collet piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.

Ball Check Valve Fails to Close Passage to Vessel Ports

This failure depends on the ball check valve sealing the passage to the vessel ports during the up-signal portion of the rod withdrawal cycle and the collet remaining unlatched. This is normal withdrawal sequence. Then, if the ball were to move up and foreign material were to jam the ball in the ball cage or prevent it from reseating at the bottom on the seat surface, water below the drive piston would return to the reactor through the vessel ports and the annulus between the drive and the housing. Because the flow resistance of this return path would be lower than normal, the calculated withdrawal speed would be 2 fps. During withdrawal, differential pressure across the collet piston would be approximately 40 psi. Therefore, the collet would tend to latch and would have to stick open before continuous withdrawal, at 2 fps, could occur. Water would flow upward past the velocity limiter piston, generating a small retarding force of approximately 120 lb.

Hydraulic Control Unit Valve Failures

Various failures of the valves in the hydraulic control unit can be postulated, but none could produce differential pressures approaching those described in the preceding paragraphs, and none alone could produce a high-velocity withdrawal. Leakage through either one or both of the scram valves produces a pressure that tends to insert the control rod rather than to withdraw it. If the pressure in the scram discharge volume should exceed reactor pressure following a

scram, a check valve in the line to the scram discharge header prevents this pressure from operating the drive mechanisms.

Collet Fingers Fail to Latch

When the drive withdraw signal is removed, the drive continues to withdraw at a fraction of normal speed. Without some initiating signal, there is no known means for the collet fingers to become unlocked. If the withdrawal drive valve fails to close following a rod withdrawal, it would have the same effect as a failure of the collet fingers to latch in the index tube. Because the collet fingers remain locked until they are unloaded, accidental opening of the withdrawal drive valve normally does not unlock them.

Withdrawal Speed Control Valve Failure

Normal withdrawal speed is determined by differential pressures in the drive and is set for a nominal value of 3 in./sec. Withdrawal speed is maintained by the pressure regulating system and is independent of reactor vessel pressure. Tests have shown that accidental opening of the speed control valve to the full-open position produces a velocity of approximately 6 in./sec.

The CRD system prevents rod withdrawal, and it has been shown above that only multiple failures in a drive unit and in its control unit could cause an unplanned rod withdrawal.

CRD Collet Retainer Tube Cracking

Two possible failure modes exist that may briefly result in an inoperable control rod drive during plant operation. These failure modes are a stuck rod at any operating condition and an unlatched rod at cold, unpressurized conditions.

In the case of a scram with a stuck rod from any hot or cold operating condition, the failure of a single rod to insert will not prevent the reactor from reaching a subcritical condition. This is because of the Technical Specification requirement that the reactor be capable of being made subcritical with the strongest rod assumed not to move from its withdrawn position. In the unlikely case that a drive adjacent to the stuck rod fails concurrently, the standby liquid control system may be required to reach a subcritical condition. For this condition to occur, the stuck drive must have occurred subsequent to the last CRD test. This must then be followed within 1 week or less by the second CRD failure, and these two events must occur at adjacent locations in the core. The resultant probability of these combined events is exceedingly remote; however, the consequences of this event can be mitigated by the operation of the standby liquid control system.

An unlatched drive condition would occur only when a drive with a failed collet retainer tube was given a high-pressure withdrawal signal in a cold depressurized vessel. The failure of a control rod drive to remain inserted subsequent to a scram (at cold critical conditions) would not prevent the return of the reactor in a subcritical condition. The consequences of a concurrent failure of a second drive would be the same as those postulated for a stuck rod with a second drive failure at the time of a scram from a critical condition. Also, a rod drift caused by this

postulated failure (from a cold critical condition) is even less severe than a rod withdrawal error occurring in the power range.

The control rod drop accident cannot occur as a result of either a step or an unlatched drifting drive since neither of these failures involve the uncoupling of the drive and the control blade.

The postulated failure of a CRD collet retainer tube cannot affect the performance of other safety systems.

4.6.2.3 Scram Discharge Volume System

The scram discharge volume system is provided to receive and contain the water exhausted from all control rod drives during a reactor scram, thereby limiting the loss of water from the reactor vessel. See also Section 3.9.4.1.2. During normal reactor operation, the discharge volume is empty and its vent valves, [redacted] and drain valves, [redacted] remain open. During a scram, the venting of the scram valve pilot header via the backup scram valves [redacted] or the venting of the air line to the vent and drain valves via solenoid valves [redacted] and [redacted] causes these valves to close. On ATWS-ARI actuation, the scram valve pilot air header is vented as described in section 3.9.4.1.2, causing the scram discharge volume vent and drain valves to close. The vent and drain valves may also be manually controlled from the control room via hand switch [redacted] and scram valves [redacted].

The scram discharge volume system consists of 8-in. diameter header piping that connects to each hydraulic control unit (0.75- in. drive water discharge line) and drains into 10-in.-diameter instrument volumes. The header piping is sized to contain the water volume discharged from all control rod drives during a reactor scram (3.34 gal per control rod drive) independent of the instrument volumes. One-half (44) of the 89 hydraulic control units are connected to header piping that drains into instrument volume "B" on the north side of the DAEC reactor building. The other half (45) are connected to scram discharge volume header piping that drains into instrument volume "A" on the south side of the building. The two instrument volumes communicate with each other via a 2-in.-diameter line connected to the bottom of each instrument volume.

The DAEC has two instrument volumes (9.5 in. inside diameter), each with scram switches that are adjacent to and below their respective scram discharge headers (7.6 in. inside diameter) and are connected to the headers with piping of diameter equal to that of the headers (7.6 in. inside diameter). This relatively short, full-diameter connection and continuous slope downward toward each instrument volume provides hydraulically immediate communication and precludes any possibility for loop seals or flow restriction/blockage to the instrument volume. It is therefore considered that each header/instrument volume configuration constitutes a single volume for the purpose of scram discharge volume water-level monitoring.

Thermally activated and float-type liquid level switches are connected to both instrument volumes to continuously monitor for abnormal water accumulation. An instrument volume high-level switch [redacted] actuates a control room alarm to indicate that the volume is not completely empty during postscram draining or to indicate that the volume starts to fill through water accumulation at other times during reactor operation. A rod-withdraw-block level switch [redacted] actuates a control room alarm and prevents further control rod withdrawal when water accumulates to half the capacity of the instrument volumes.

Eight additional switches are interconnected with the trip channels (A-1, A-2, B-1, B-2) of the reactor protection system and will initiate a reactor scram should water fill the instrument volumes. Four switches are mounted on each instrument volume: the A-1 and B-1 trip channel switches on the A instrument volume and the A-2 and B-2 trip channel switches on the B instrument volume. The switch logic is arranged such that a reactor scram will occur when either A trip channel and either B trip channel is activated (one-out-of-two-twice logic). Therefore, a scram will be initiated if both an A trip channel switch and a B trip channel switch on either instrument volume are tripped or if an A trip channel switch on one instrument volume and a B trip channel switch on the other instrument volume are tripped.

The required prescram, free volume in the scram discharge volume system for the DAEC is (3.34 gal per control rod drive times 89 control rod drives) 300 gal. The available volume above the instrument volumes is in excess of 400 gal.

Approximately 30 representative line slope checks were performed throughout the scram discharge volume system, including the headers, drain piping, and vent piping. In all cases, slopes were in the proper direction (that is, slope up toward the vent valve and down toward the drain valve). This provided assurance that essentially no water is held in the 8-in.-diameter headers or anywhere else in the system when the system is gravity drained.

An investigation was performed to determine whether water could enter the scram discharge volume headers or instrument volumes via either the vent or drain lines for the system. The vent piping outside of the vent valves is a dedicated line open to the reactor building atmosphere. No mechanism for drawing or forcing water into this line could be identified, given this piping configuration. The 2-in. drain line for the system enlarges to a 4-in. drain header downstream of the drain valves. The 4-in. drain header, before terminating above water level in the equipment drain sump, can accept drainage via two other connections: (1) a 3-in. drain line from the condensate storage tank system (tank drain, overflow tank overflow, and sample pit drain) and (2) a 2-in. instrument condensate drain line from an offgas system panel, [redacted]. If the 4-in. drain header to the sump is open (not plugged), neither of these two water sources has sufficient pressure and water volume to cause water to back up the 4-in. drain header, through the scram discharge volume drain valve, and up into the instrument volumes. Concerning the possible plugging of the 4-in. line between these connection points and the end of the line at the sump (thereby causing water from either source to back up toward the scram discharge volume), no mechanism for such plugging could be identified because only lines smaller than 4 in. (that is, 2 and 3 in.) discharge into this 4-in. header, and the header terminates above sump water level. Therefore, it is concluded that water will not enter the scram discharge volume, thereby reducing

the available free volume required to perform its safety (scram) function, by either the vent line or the drain line.

The scram discharge volume vent piping vents the entire scram discharge volume system through two vent valves, and the vent valve discharge is open to the reactor building atmosphere via a dedicated line. From the standpoints of both system safety function (scram) and draining the system after scram reset, this configuration is adequate.

4.6.2.4 Rod Worth Minimizer System

Control rod withdrawal sequences developed limit the reactivity worths of control rods in the core, and together with the integral rod velocity limiters, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/g (see Reference 6). The peak fuel energy content of 280 cal/g is below the energy content at which rapid fuel dispersal and primary system damage are assumed to occur.

The Rod Worth Minimizer system will prevent the operator from inadvertently selecting and moving a high worth rod in the startup and low-power ranges. Above 10% power, the results
of the rod drop accident with the worst single operator error are less than 280 cal/g. Therefore, this system, in addition to normal operating procedures, prevents the postulated rod drop accident from exceeding 280 cal/g over the entire range of plant operating conditions and core exposure.

In the event that the Rod Worth Minimizer is out of service, when needed, a second licensed operator or other qualified member of the technical staff can manually fulfill the control rod pattern conformance functions of the Rod Worth Minimizer.

4.6.3 TESTING AND VERIFICATION OF THE CRD SYSTEM

4.6.3.1 Measured Scram Times

The following three methods of data collection were used at the DAEC to obtain measured scram time data:

1. Process computer scram time data.
2. Digital computer data.
3. Individual control rod scram time data.

Type 1 and type 2 scram time data collection methods were used at DAEC previously. The usual method of scram data collection is type 3; recording of individual control rod scram time data.

These may be differentiated by the size of the given sample as follows:

1. A sample size of 19 or less is process computer data.
2. A sample size of 44 is digital computer data.
3. A sample size of 89 is individual rod scram test data.

The type 3 data (full-core tests or individual rod scram data) on which the DAEC analysis was based are scram tests of all drives in the core (tested individually) normally conducted when a plant initially starts up and as part of each refueling outage thereafter. The tests are conducted at or above 800 psig reactor pressure with a fully charged accumulator. The CRD system water supply pump is isolated so as not to affect the individual drive scram time. A full-core test can be conducted as part of a vessel hydro test or while the plant is being brought up to power.

The type 1 data (process computer scram time data) were obtained by instrumenting a number of drives (maximum of 19 at the DAEC) and using the plant process computer to record scram time data during actual scrams. The type 2 data were obtained in a manner similar to the type 1 data except that an auxiliary computer was installed at the DAEC. This extra computer power allowed 44 rods (all those full out during normal operation) to be instrumented. Only one scram occurred while this computer was installed at the DAEC.

4.6.3.1.1 Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent exceeding the safety design bases contained in Chapter 15.

In the analytical treatment of the transients, 390 msec are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typically observed time delay of about 270 msec. The 200 msec time is included in the allowable scram insertion times specified in the Technical Specifications. In addition, the control rod drop accident has been analyzed in NEDO-10527[6] and its Supplements 1 and 2 for the scram times given in the Technical Specifications. After each refueling outage, all operable control rods are scram tested individually for allowable insertion times from the fully withdrawn position, with the nuclear system pressure at or above 800 psig and before the 40% power level is reached. Surveillance requirements for the control rod scram times are contained in the Technical Specifications.

4.6.3.1.2 Reactivity Anomalies

During each fuel cycle, excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous
2018-005 behavior in the excess reactivity may be detected by a comparison of the monitored core k_{eff} at selected base states to the predicted core k_{eff} at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. In addition, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency ensures that a comparison will be made before the core reactivity change exceeds 1% $\Delta k/k$. Deviations in core reactivity greater than 1% $\Delta k/k$ are not expected and require thorough evaluation. One percent reactivity limit is considered safe because an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

4.6.3.2 CRD Operating and Testing Procedures

CRD operating instructions are contained in the DAEC operating instructions. The surveillance test procedures for the CRD system include the following:

1. Control rod exercise.
2. Scram insertion times.
3. Refueling interlocks functional tests.
4. Control rod maintenance reactivity margin test.
5. CRD housing support inspection.

In addition, the integrated plant operating instructions, surveillance test procedures, and NGD procedures cover operation during control rod testing.

4.6.4 INFORMATION FOR COMBINED PERFORMANCE OF REACTIVITY SYSTEMS

4.6.4.1 Vulnerability to Common Mode Failures

The two reactivity control systems, control rod drive and standby liquid control, do not share any instrumentation or components. Thus, a common mode failure of the reactivity systems would be limited to an accident event which could damage essential equipment in the two independent systems.

A seismic event or the postulated accident environments are not considered potential common mode failures since the essential (scram) portions of the CRD system are designed to Seismic Category I standards and to operate as required under postulated accident environmental conditions. The SLC system is also designed to Seismic Category I standards.

No common mode power failure is considered possible. The scram function of the CRD system is "fail-safe" on a loss of power and is designed to override any other CRD function. The SLC system has two independent power supplies to its essential redundant pumps and valves. The power supplies to the SLC system are considered vital and as such are switched to the on-site standby diesels on a loss of normal power sources.

4.6.4.2 Accidents Taking Credit for Multiple Reactivity Systems

There are no postulated accidents evaluated in Chapter 15 that take credit for two or more reactivity control systems preventing or mitigating each accident.

4.6.4.3 Evaluation of Combined Performance

As indicated in Section 4.6.4.2, credit is not taken for multiple reactivity control systems for any postulated accidents in Chapter 15.

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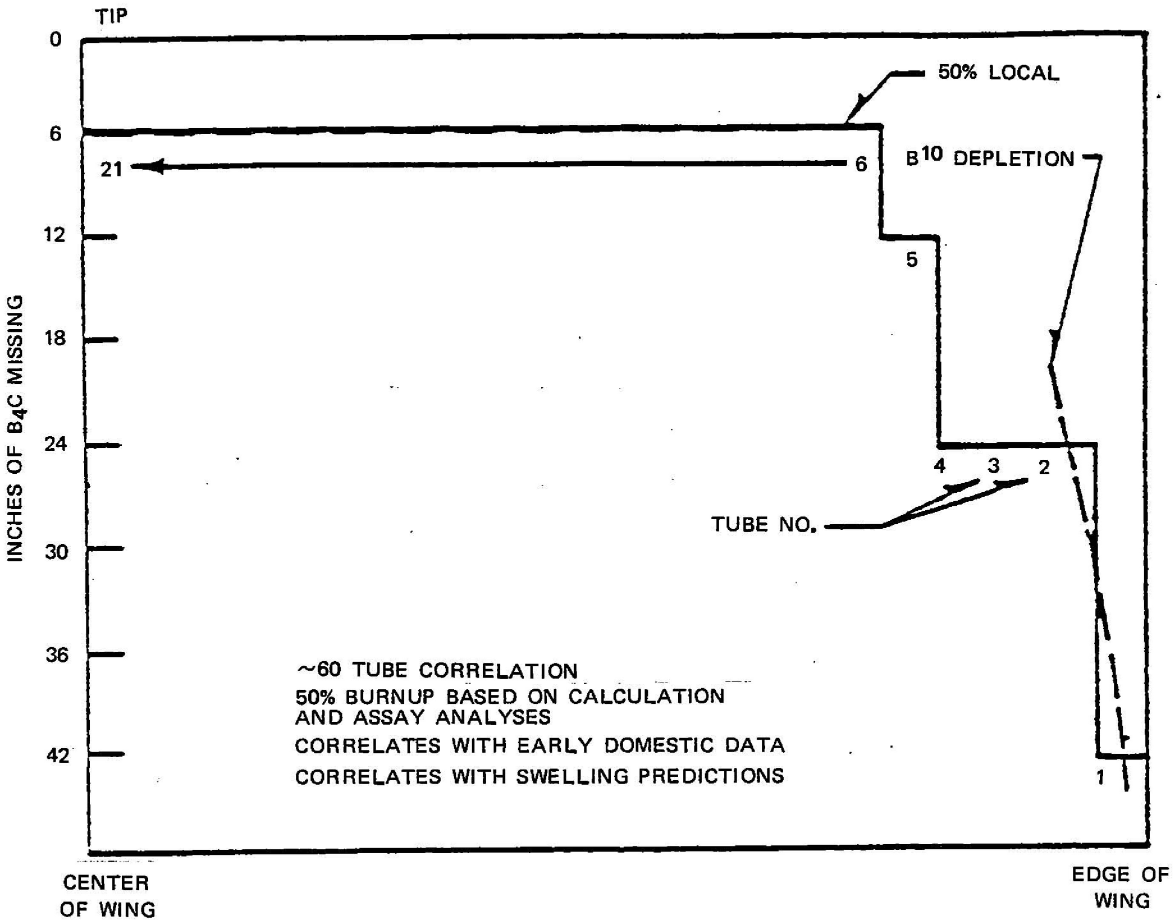

DUANE ARNOLD ENERGY CENTER
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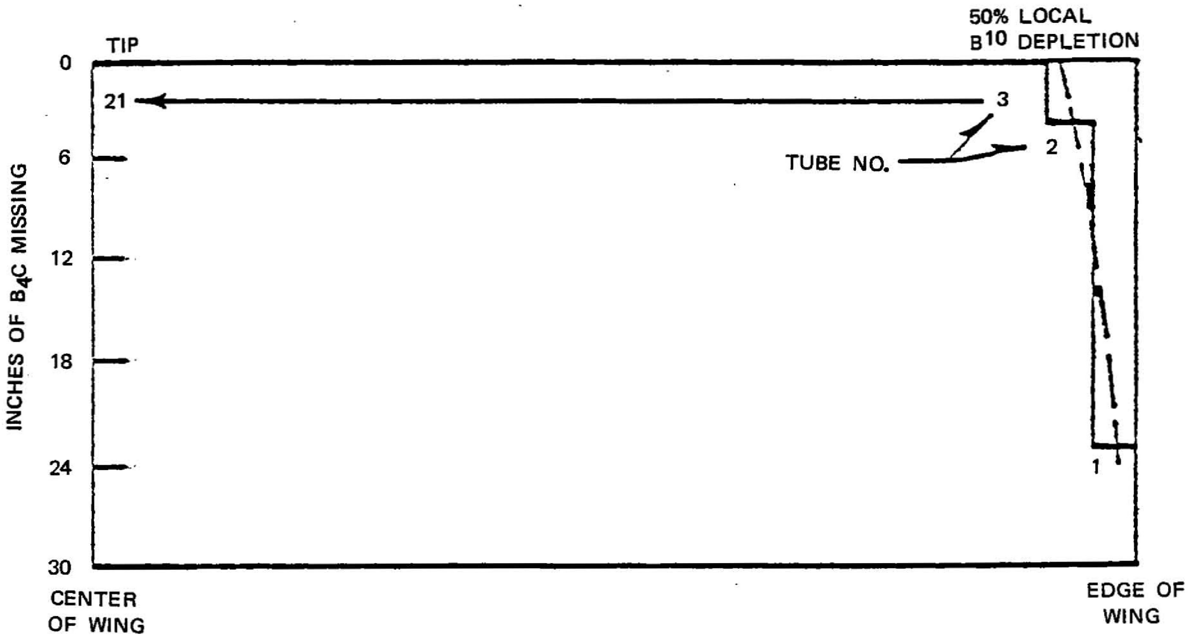
UPDATED FINAL SAFETY ANALYSIS REPORT

Loss Versus Depletion Correlation
(100% Life)

Figure 4.6-3



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(80% Life)

Figure 4.6-4