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July 13, 1984

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Mr. D.G. Eisenhut, Director
Division of Licensing

SUBJECT: IN THE MATTER OF 238 NUCLEAR ISLAND GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT (GESSAR II)
DOCKET NO. STN 50-447

NUCLEAR ISLAND/BOP INTERFACES AND GESSAR II EVOLUTION

Reference: Memorandum for C.O. Thomas (NRC) from D.C. Scaletti (NRC), "GESSAR II Meeting Summary," June 7, 1984

In the reference memorandum, closure activities for the GESSAR II severe accident review were enumerated. Items requiring additional General Electric input included documentation of Nuclear Island/Balance of Plant Interfaces, including interface assumptions in the Probabilistic Risk Assessment (PRA), and documentation of the GESSAR II design evolution.

Attachment 1 describes how the Nuclear Island (NI)/Balance of Plant (BOP) interfaces are controlled in the GESSAR II design. The non-severe accident or FDA-1 aspect of NI/BOP interface is covered in detail in Section 1.10 of the GESSAR II SER. In summary, the Staff determined that the interface requirements provided in GESSAR II are adequately descriptive to ensure the compatibility of the GESSAR II design with the BOP designs that would be submitted in individual applications referencing GESSAR II.

In keeping with FDA-1 and to assure reliability objectives are met, the key PRA interfaces are also included as NI/BOP interfaces. To further assure that reliability objectives are met during procurement, construction, preoperational testing, startup testing, operation and maintenance, GESSAR II Chapter 17 (Quality Assurance) has been modified to require that the Applicant's performance specifications and monitoring procedures include these key interface requirements.

Attachment 2 is the report documenting the evolution of the GESSAR II design. It begins with GE's choice of the BWR as its design basis with a NSSS scope of supply in 1955 to the 1984 GESSAR II BWR/6 Mark III design with a Nuclear Island scope of supply. The evolution presented

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is in two general categories. The first category addresses the design evolution and focuses on the major components: reactor, containment and the Nuclear Island itself. The second category addresses evolution through experience and testing. This latter category includes operational feedback, abnormal occurrences and testing.

If there are any questions on the information provided herein, please contact me or K.W. Holtzclaw (408) 925-2506 or J.N. Fox (408) 925-5039.

Very truly yours,

W.H. D'Ardenne for J.F. Quirk

Joseph F. Quirk, Manager
BWR Systems Licensing
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Attachments

cc: F.J. Miraglia (NRC)
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C.O. Thomas (NRC)
L.S. Gifford (GE-Bethesda)
R. Villa (GE)

CONTROL OF NUCLEAR ISLAND/BALANCE OF PLANT INTERFACES

Background

The Nuclear Island (NI)/balance of plant (BOP) interfaces are illustrated in Figure 1. The complex interfaces that existed between the Nuclear Steam Supply System (NSSS) and BOP when the GE scope of supply was limited to the NSSS are practically eliminated by the NI scope. The only remaining BOP interface areas are those between the NI and Turbine Island (TI) and between the NI and service facilities. This change in scope reduces the magnitude of interfaces from tens of thousands to no more than hundreds.

Interfaces

In GESSAR II, the interfaces are classified as either GESSAR II/FSAR interfaces (GESSAR II, Tables 1.9-1 through 1.9-19) or NI/BOP design interfaces (GESSAR II, Tables 1.9-20 through 1.9-23 and Figures 1.9-1 through 1.9-5). The GESSAR II/FSAR interfaces fall into one of the following five categories:

1. BOP scope (difference between Regulatory Guide 1.70 and NI scope)
2. Equipment vendor dependent
3. Applicant dependent
4. Site dependent
5. Deferred until first Applicant references GESSAR II

Strictly speaking, only the Category 1 GESSAR II/FSAR interfaces and NI/BOP design interfaces are "NI/BOP Interfaces." However, GE chose to include all of the interfaces as NI/BOP interfaces to assure that the Applicant will provide compatible design features and meet reliability objectives.

As a final measure in meeting reliability objectives, the key probability risk assessment (PRA) interfaces shown in new GESSAR II Table 1.9-24 are included as NI/BOP interfaces. The specific PRA interfaces in this table are the result of a review of the PRA assumptions (such as reliability or operability assumptions) and an exclusion of those assumptions which met one or more of the following:

1. Characteristics well defined by the GESSAR II design documentation.

2. Recognized industry data base (component reliability or operator action time).

3. Little importance to the PRA conclusions (e.g., 100% change in reliability changes the corresponding overall PRA results by less than 1%).

Any borderline or questionable PRA interface was retained as a key PRA interface.

Control

The levels of requirements imposed on the Applicant are the same ones used by GE for the design of the NI proper, and GE has formal documentation in place to control these NI/BOP interfaces. General Electric assures compliance by periodically reviewing all of the interfaces by GE teams that visit the Applicant. The Applicant audits his own AE (architect engineer) which provides further verification of conformance to the interface requirements. The AE is also subject to independent QA verifications within his own in-house procedures. General Electric ensures compliance with the NRC licensing reviews by verifying that the NI/BOP interfaces are met.

To further assure that reliability objectives are met during procurement, construction, preoperational testing, startup testing, operations and maintenance, GESSAR II Table 1.9-17, Subsection 17.1.2 and Section 17.2 will be modified as indicated to require that the Applicant's performance specifications and monitoring procedures include the applicable interface requirements of Tables 1.9-1 through 1.9-24 and of Figures 1.9-1 through 1.9-5. For completeness, these GESSAR II Chapter 17 interface requirements were also added to Table 1.9-17.

In terms of construction controls, GE has procedures to control the interfaces and has approval of non-conformance and approval of as-built documentations.

Site controls are provided by the licensing process itself in that the GESSAR II SER (NUREG-0979) requires the NRC to perform a site specific review of the reference plant site characteristics (meteorology, hydrology and seismology) to demonstrate that they are compatible with the GESSAR II siting envelope assumptions (GESSAR II, Table 2.0-1).

NRC Review

Some of the major systems that were reviewed for interface consistency by the Staff are the fuel oil, the essential service water supply, instrumentation controls, condensate storage, power feeders, liquid radwaste, fire protection and feedwater.

As noted in the GESSAR II SER, the NRC's review and evaluation addressed the interface requirements either from the standpoint of general design provisions (i.e., qualitative); specific design provisions (i.e., quantitative), by incorporation or reference to the interface requirements in GESSAR II; or, by a description of the interface mechanism between GESSAR II and the BOP.

Also noted in the GESSAR II SER is that the NRC audited detailed interface information that is supplied to reference plant applicants. The NRC acknowledged that this interface information has always been part of the contractual arrangements between the NSSS designer and the BOP designer; however, for the purpose of a standard NI design, the safety-related, interface requirements are significant for reference by Applicants in the future.

Finally, as noted in the GESSAR II SER, the NRC determined that the interface requirements provided in GESSAR II are adequately descriptive to ensure compatibility of the GESSAR II design with the BOP designs that would be submitted in individual applications referencing GESSAR II.

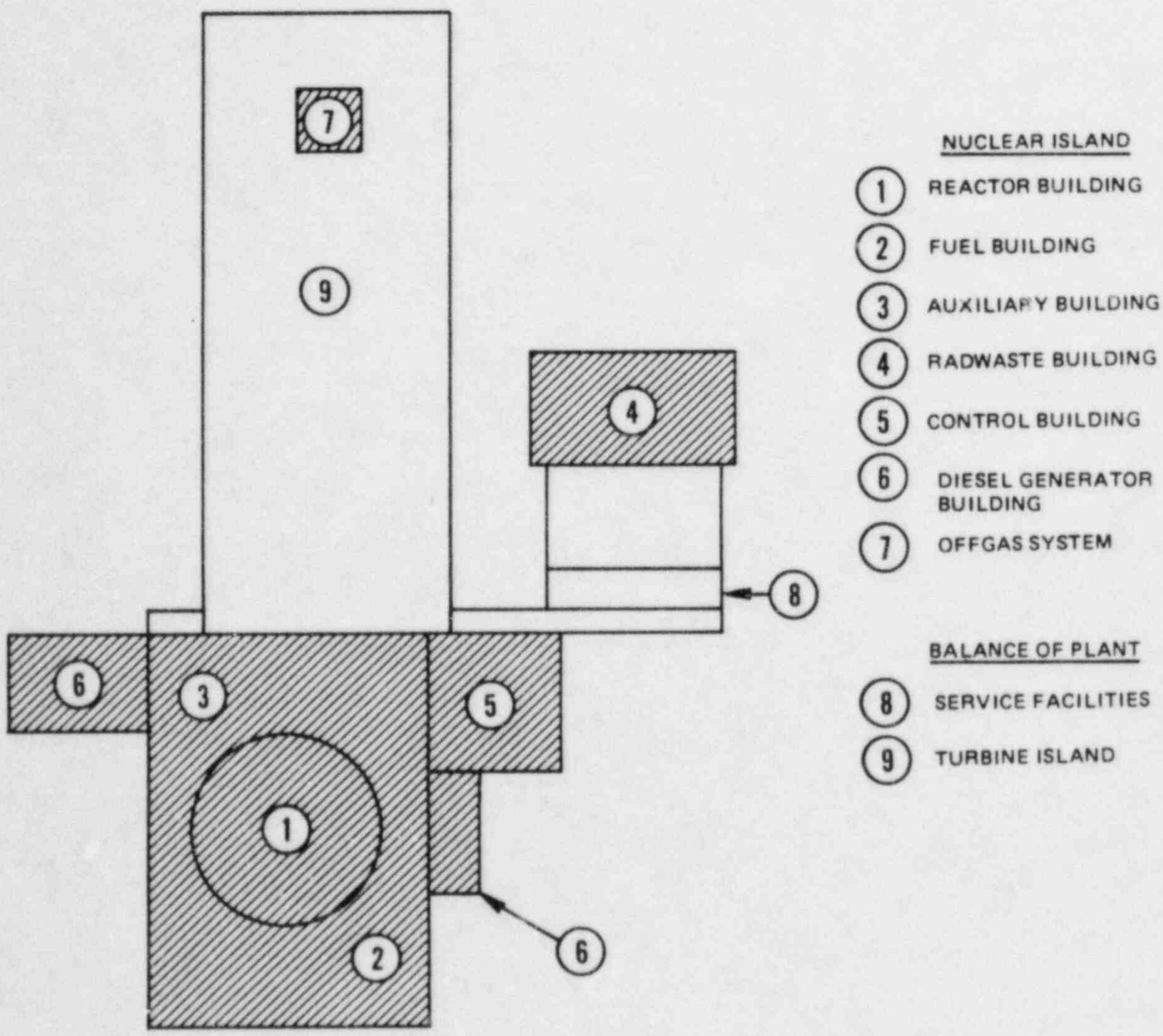

Figure 1. Nuclear Island/Balance of Plant Interfaces

1.9.1.3 Nuclear Island-BOP Design Interfaces (Continued)

allowables are specified as acceptable. Interface control documentation is provided which will indicate the exceptions.

1.9.1.4 PRA Interfaces

The key PRA interfaces and interface requirements are provided in Table 1.9-24. The Applicant will demonstrate that the BOP design is consistent with these interface requirements before applying the PRA results of Section 15D.3 to his FSAR. If not consistent, the Applicant must demonstrate that there is a negligible impact on the overall public risk.

1.9.2 Exceptions

Applicant will supply.

1.9.3 References

1. Letter, J. F. Quirk to D. G. Eisenhut, "GESSAR II Seismic Event Analysis," September 21, 1983.

2. Letter, J. F. Quirk to D. G. Eisenhut, "Information in Response to Request for Additional Information Regarding GESSAR II Severe Accidents," January 31, 1984.

17.2 QUALITY ASSURANCE DURING THE OPERATIONS PHASE

The Applicant's performance specifications and monitoring procedures will include the applicable interface requirements of Tables 1.9-1 through 1.9-24 and of Figures 1.9-1 through 1.9-5 to assure reliability objectives are met and to prevent degradation of the reliability during operation and maintenance.

The remainder of this section will be provided by the Applicant.

17.1 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION

17.1.1 Organization

See Section 1 of Reference 1.

17.1.2 Quality Assurance Program

The identification of safety-related structures, systems, and components (Q-list) to be controlled by the quality assurance program is the responsibility of the Applicant. The Applicant will supplement and clarify its Q-list in accordance with Question 17.3. The appropriate items will be added to Table 3.2-1. The reamining items will be subject to the pertinent requirements of GE's and/or the Applicant's QA programs unless otherwise justified. 17.3

The Applicant's performance specifications and monitoring procedures will include the applicable interface requirements of Tables 1.9-1 through 1.9-24 and of Figures 1.9-1 through 1.9-5 to assure that reliability objectives are met during procurement, construction, preoperational testing, startup testing and the formulation of procedures for operations and maintenance.

The remainder of this subsection is covered in Section 2 of Reference 1.

17.1.3 Design Control

See Section 3 of Reference 1.

17.1.4 Procurement Document Control

See Section 4 of Reference 1.

17.1.5 Instructions, Procedures, and Drawings

See Section 5 of Reference 1.

17.1.6 Document Control

See Section 6 of Reference 1.

17.1.7 Control of Purchased Material, Equipment, and Services

See Section 7 of Reference 1.

Table 1.9-24

PRA INTERFACES

Item No.	Subject	GESSAR II/FSAR Interface Table No.	GESSAR II/FSAR Interface Item No.	GESSAR II/FSAR Interface Subsection	PRA Interface Requirement
1.	Grid Reliability Analysis	1.9-8	8.6	8.2.2	Initiation Frequency <0.05 events/year and loss of feeder probability $<10^{-2}$ (Table D2-14 of Appendix D to Section 15D.3)
2.	ESW Reliability Analysis	1.9-9	9.6	9.2.1	Sufficient as to not degrade the conclusions in Appendix D of Section 15D.3 tables: D2-2 RCIC D2-11 ESW to RHR/LPCS D2-14 EDG Service Water
3.	Seismic Hazard Curve, Geology and Seismology	1.9-2	2.28	2.5.1	Site hazard curve response within Figure 2-1 of Reference 1. Geology and seismology same as GESSAR II/ FSAR interface.
4.	Meteorology	1.9-2	2.10 and 2.11	2.3.4 and 2.3.5	Total risk within Figure 7.1-2 of Section 15D.3 bounds. Site unique data to be applied to confirm applicability of risk conclusions.
5.	Population Distribution	1.9-2	2.3	2.1.3	
6.	Emergency Planning	1.9-1	1.39	1.8.101	

Table 1.9-24

PRA INTERFACES (Continued)

Item No.	Subject	GESSAR II/FSAR Interface Table No.	Item No.	Subsection	PRA Interface Requirement
7.	Containment Design	1.9-3	3.24	Table 3.8-3	Failure location and capability consistent with Appendix G of Section 15D.3.
8.	Emergency Procedures	1.9-1	1.68	1A.8	Plant emergency procedures consistent with EPGs.
9.	Maintenance Procedures	1.9-1	1.27	1.8.33	Consistent with Reference 2: a. References 3 and 4 of Tables D.2.1-1 and D.2.4-1 b. Footnote 1 of Table 2.2.3-1
10.	Flood and Groundwater	1.9-2	2.16 and 2.26	2.4.3 and 2.4.13	Same as GESSAR II/FSAR interface.
11.	Ultimate Heat Sink	1.9-9	9.10	9.2.5	Same as GESSAR II/FSAR interface.
12.	Site-Dependent Blasts	1.9-2	2.6	2.2.3.1	Same as GESSAR II/FSAR interface.
13.	Collapse of Non-Seismic Category I Components	1.9-3	3.5	3.3.2.3	Consistent with Reference 1, Table 3-18.

Table 1.9-24

PRA INTERFACES (Continued)

Item No.	Subject	GESSAR II/FSAR Interface			PRA Interface Requirement
		Table No.	Item No.	Subsection	
14.	Missiles Generated by Natural Phenomena	---	---	---	Same as Subsection 3.5.1.4 requirement.
15.	Turbine Missiles	1.9-3	3.9	3.5.1.3.4	Same as GESSAR II/FSAR interface requirement.
16.	Aircraft Hazards	---	---	---	Same as Subsection 2.2.2.5 requirement.

Table 1.9-17

CHAPTER 17
GESSAR II/FSAR INTERFACES

ITEM NO.	SUBJECT	DESCRIPTION	PAGE	SUBSECTION	RELATED QUESTION	INTERFACE CATEGORY
17.1	Q-List	Identify safety-related structures, and components (Q-List) to be controlled by the quality assurance program. Performance specifications and monitoring procedures to include the applicable interface requirements of Tables 1.9-1 through 1.9-24 and Figures 1.9-1 through 1.9-5 in performance specifications and monitoring procedures.	17.1-1	17.1.2		3
17.2	QA During the Operating Phase	Describe the QA program that will assure the quality of all safety-related items and activities during the operations phase per R.G. 1.70 Section 17.2. Performance specifications and monitoring procedures to include the applicable interface requirements of Tables 1.9-1 through 1.9-24 and Figures 1.9-1 through 1.9-5 in performance specifications and monitoring procedures.	17.2-1	17.2		3

EVOLUTION OF THE GENERAL ELECTRIC
GESSAR II BWR/6 NUCLEAR ISLAND DESIGN

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TABLES

ILLUSTRATIONS

1. INTRODUCTION

In 1955 General Electric Company (GE) responded to President Eisenhower's "Atoms for Peace" program by embarking on an arduous technical program to develop a large central-station nuclear power plant. The first result of that early developmental effort was the 5 MWe Vallecitos Boiling Water Reactor (VBWR) - The first licensed power reactor in the United States and connected into the electrical grid of Pacific Gas and Electric Co. (PG&E).

Now, almost 30 years later, the GE Boiling Water Reactor (BWR) has evolved into a major contributor to our nations electrical power supply system using the same attractive direct-cycle features (see Figure 1-1) which prompted GE to select this design approach from the outset. The purpose of this report is to describe these evolutionary changes, and to provide insights affecting decisions to make the changes.

Early light water reactor (LWR) technology was developed principally through the U.S. Navy program for ship propulsion. That program focused on the indirect-cycle Pressurized Water Reactor (PWR) technology. General Electric participated in the PWR program principally through its activities conducted for the Navy at the Knolls Atomic Power Laboratory in Schenectady, New York. The successful operation of early naval reactors caused industry to give serious attention to the possibility of commercial LWRs. The fact that the underlying technology of steam power plants was familiar and well developed gave industry the courage to aggressively undertake the commercialization of the LWR concept. Although several industry participants chose the PWR as the basis for its commercial power plant designs, GE chose the BWR as its design basis.

1.1 DIRECT CYCLE ADVANTAGES

The direct cycle BWR offered the following attractive technological advantages which affected that decision:

1. Simplicity - The single vessel and direct reactor-to-turbine cycle is a simple nuclear approach for converting steam to electricity. With no pressurizer or steam generators, the direct cycle design has fewer large components than the indirect cycle.

2. Lower Pressure - BWRs typically operate near 1000 psi, about one-half the pressure of other LWR types. Lower pressures allow operation at the saturation conditions of the coolant, less potential for leakage of reactor water and easier delivery of normal and emergency water to the reactor vessel.

3. Strong Negative Void Coefficient - The existence of steam voids in the BWR provides an inherent check on excessive power excursions. In the event of an unwanted increase in reactivity, the volume of steam voids increases, thereby reducing the quantity of moderator in the core and inhibiting the nuclear reaction.

4. Coupling Between Power and Core Flow - Because of boiling in the core, steam voids, which significantly influence fission rate, can be controlled by core flow. Therefore, reactor power changes can be achieved by merely varying reactor coolant recirculation flow without moving control rods.

5. Water Sources Directly to Reactor - The direct cycle permits BWR normal and emergency water delivery systems to feed directly into the reactor vessel where they can be used to protect the core in the event of an emergency.

1.2 DIRECT CYCLE CHALLENGE

General Electric believed that the above features would lead to overall advantages in safety, economics, reliability, control, and maneuverability. At the same time, GE recognized that development of the direct-cycle BWR involved a number of technical challenges. Some of these were evident at the beginning - others were not. Three of these challenges merit further discussion here:

1. Boiling Technology - The development of the BWR involved the coupling of neutronic, thermal and hydraulic phenomena within the reactor core. Methods for analyzing steady-state and transient behavior, and for ensuring stability, had to be developed and qualified.

2. Radioactivity Carryover - The direct-cycle BWR involved potential carryover of radioactivity to the turbine. It had to be demonstrated that such carryover would not unduly restrict turbine cycle operation and maintenance.

3. Direct-Cycle Water Chemistry - The direct-cycle BWR circulated turbine cycle water through the reactor and, therefore, involved potential water-chemistry and materials-related issues. It had to be shown that such issues were manageable.

Thorough investigation of each of these issues has been required over the years to fully qualify the direct-cycle approach.

1.2.1 Boiling Technology

The direct-cycle reactor system allows coolant boiling inside the reactor vessel. The resulting two-phase condition involves the coupling of neutronic and thermal-hydraulic phenomena within the reactor core. There were early skeptics who said that the characteristics of boiling water in a reactor core would prevent the generation of large quantities of power in a stable, controlled manner. Today, the practicality of BWR operation is demonstrated by over 50 GE-type BWRs operating worldwide.

Boiling technology has evolved and been refined, piece by piece, over the years. Analytical models were developed that could analyze the coupled neutronic and thermal-hydraulic phenomena in the reactor vessel and accurately predict in-core behavior. The Atlas test facility (see Section 8.13) was built by GE to perform testing on full-scale electrically-heated BWR fuel bundles to study and understand the thermal-hydraulic phenomena that take place within a BWR fuel bundle. Fuel bundle channels, a standard BWR feature, allow testing and modeling of a single bundle, with additive application of these single bundle results to the analysis of an entire reactor core.

To confirm laboratory testing and analysis, extensive fullscale tests in operating BWRs have been run. Calculated and measured transient characteristics have been compared. Today, qualified models are in place to analyze the coupled neutronic, thermal and hydraulic phenomena in the BWR core. The combination of such analytical and experimental efforts has provided a very solid BWR technology data base, and demonstrated that boiling in the reactor is not only feasible, but is an easily controlled and safe means of generating steam.

1.2.2 Radioactivity Carryover to Turbine

The second challenge facing the direct cycle was the potential carryover of radioactivity to the turbine. During early BWR development, some uncertainty existed as to whether fuel could be made reliable enough to avoid restrictive contamination of the turbine. Reviewing this issue today, General Electric's 1980 8x8 cumulative fuel failures are low and equivalent to less than one rod failure in every 10,000 operated. This low fuel rod failure level means that few fission products ever reach the reactor coolant, and the failure levels are getting smaller each year as even more reliable fuel designs go into service. However, even when fuel failures do occur, experience has shown that radioactivity carryover to the turbine is small and does not represent a hindrance to turbine maintenance. Most of the radioiodines and any other fission products released to the reactor water remain in the reactor vessel and are not transported with the steam to the turbine. Decontamination of the coolant at the water-steam interface in the reactor yields a decontamination factor greater than 100 for most nongaseous radioiodines, and the gaseous radioiodines are transported to the offgas system where they are removed from the steam generation cycle.

In addition, eliminating the external steam drum and employing internal steam separators/dryers resulted in improved steam quality leaving the reactor and, therefore, contributed to even further reduced carryover to the turbine.

There is some carryover to the turbine of radioactive Nitrogen-16 (N-16) formed through irradiation of reactor water. Its personnel effect is small since turbine shielding precludes significant occupational exposure during plant operation. Also, after shutdown, N-16 rapidly decays to insignificant levels (7-second half life) by the time turbine temperatures are low enough to allow maintenance to begin.

The small contribution of radioactivity carryover to the turbine on BWR occupational exposure is evident in reported data from operating plants. Typically, less than 3% of total plant occupational exposure is attributed to turbine surveillance and maintenance. In fact, BWR turbine maintenance is usually performed today by workers in coverall dress with no other special clothing or equipment. In short, BWR turbine maintenance can proceed as soon as thermal conditions permit; radiation is not the limiting factor.

1.2.3 Direct-Cycle Water Chemistry

The third technical challenge involved in the development of the direct cycle was in the area of water chemistry and materials. BWR water quality specifications were established early in the BWR development process, and condensate treatment and reactor water cleanup systems were engineered to satisfy those specifications. It was recognized early that corrosion rates of both austenitic and ferritic steels exposed to neutral high-temperature BWR water (550°F) would be small, and would not require the use of complex chemical inhibitors.

Then, almost 15 years after the start up of Dresden 1, the unexpected intergranular stress corrosion cracking (IGSCC) of welded stainless steel piping was experienced in some operating BWRs. Although the actual occurrence rate of BWR pipe cracking has been small (<1% of all welds), these failures detracted from overall GE BWR plant performance. General Electric committed its corporate resources to the resolution of the problem, and since 1975, has put in place several scientific laboratories and facilities to develop and qualify solutions. These include the free world's largest full-scale pipe test laboratory (PTL). This laboratory was built to accelerate stress corrosion conditions, duplicate field cracking, and qualify solutions to the intergranular stress corrosion cracking problem. The 72 specimen stations are capable of simultaneously testing over 1400 different weld heat-affected zones. In the presence of accelerants, reproduction of a 2-year field crack can now be accomplished in approximately 100 hours of laboratory testing.

Following 4 years of testing of over 4000 heat-affected zones, IGSCC solutions are qualified and being implemented in operating plants, plants under construction and in future BWR/6 designs. In addition to the extensive testing for use of qualified nuclear-grade stainless steels in the GESSAR II design, all other BWR nuclear system plant materials have been similarly tested and shown fully qualified for service. IGSCC is also discussed under Section 8.1.

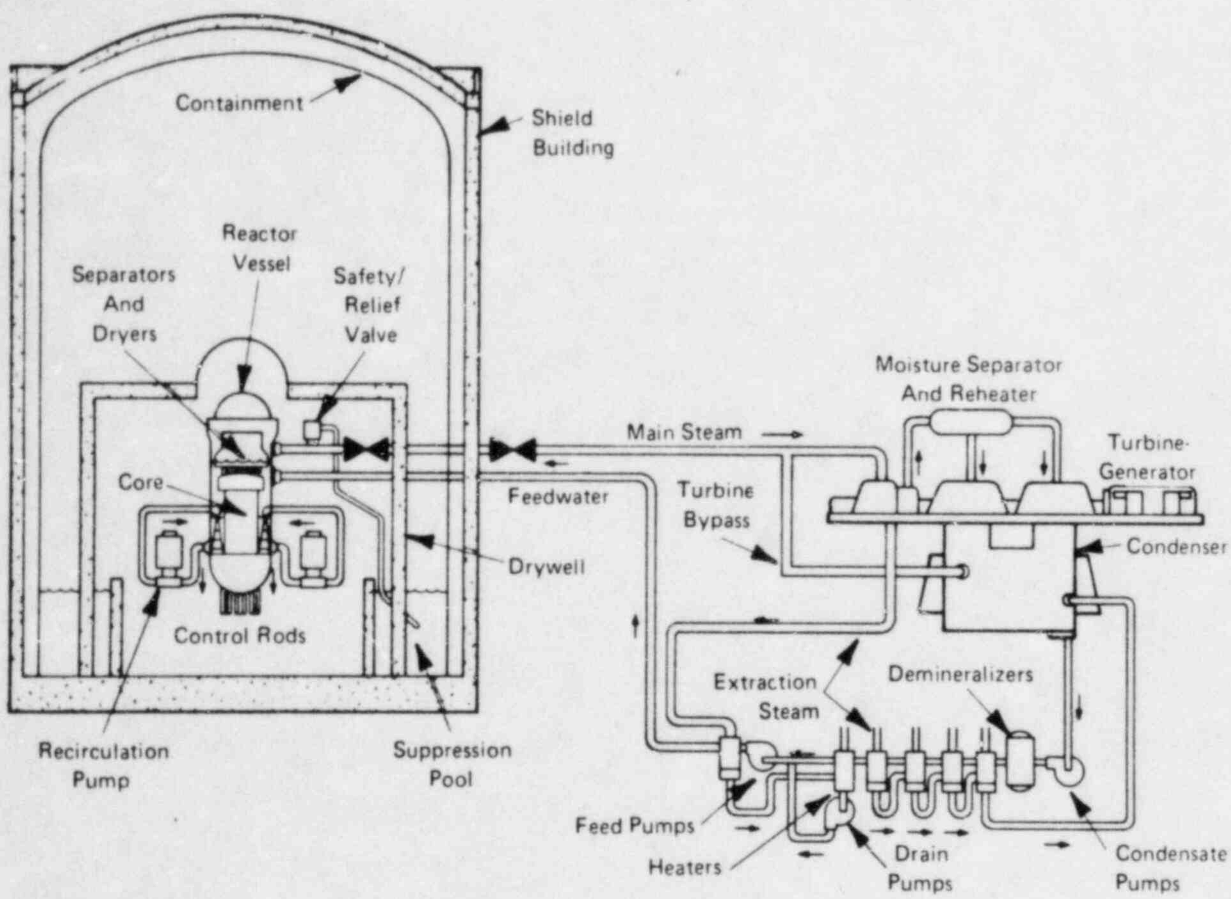


Figure 1-1. Direct-Cycle BWR Nuclear System

2. SUMMARY

The evolution of the GESSAR II BWR/6 Nuclear Island is presented in two categories. The first category addresses the design evolution and focuses on the major components: reactor, containment and Nuclear Island. The second category addresses evolution through experience and testing. This latter category includes operational feedback, abnormal occurrences and testing. Figure 2-1 provides a summary of the evolution of the GESSAR II design in terms of these two categories.

2.1 REACTOR DESIGN

General Electric embarked on the BWR with full knowledge that it was a product which would require more development and investment in technology that, in part, could have been avoided by adapting the existing U.S. Navy PWR technology to central power station. But nevertheless, GE chose to pursue the BWR to achieve the benefits in the direct cycle noted in Section 1. The early reactors and the evolutionary simplification of the design (e.g., steam generator elimination, jet pumps) are addressed in Section 3.

2.2 PRESSURE SUPPRESSION CONTAINMENT DESIGN

General Electric's second major move in developing nuclear technology was the choice of pressure suppression as its reference containment concept. Here, again, GE consciously and deliberately departed from the course that others were following. While early GE BWRs were housed in dry containments, GE saw long-term advantages in pressure suppression:

1. High Heat Capacity - Access to nearly a million gallons of water for storing large quantities of heat inside the containment was an attractive plant protection feature. With more than sufficient passive heat sink capacity to accommodate the primary system stored

energy, the operator would be able to focus his early attention on stabilizing any reactor transient without being simultaneously concerned with removing core decay heat from the containment.

2. Low Pressure - The current Mark III pressure suppression containments are designed for 15 psig - much lower than the early typical dry containment designs. Such low pressure offers construction advantages and reduces the driving force for potential fission product leakage beyond the primary containment boundary.

3. Depressurization Capability - The ability to depressurize the primary system into a self-contained heat sink provided another attractive plant protection feature. Depressurization of the primary system provides the reactor vessel with access to the low-pressure water supplies in the event high-pressure water supplies are unavailable.

4. Fission Product Retention - The suppression pool also provides "scrubbing" of fission products which might be released from the primary system. The drywell channels any such release to the pool where radioactive halogen and particulate concentrations would be reduced by a factor of at least 10^2, and perhaps as much as 10^4 before being released to the containment free volume.

The evolution of pressure suppression containment is contained in Section 4.

2.3 NUCLEAR ISLAND DESIGN

About the time GE knew it had a BWR/6 Mark III containment offering with significantly enhanced safety margins, the initial NRC policy statement on standardization was issued (April 1972). At this time GE came to the conclusion that the traditional NSSS scope was not enough and concluded that the Nuclear Island (i.e., structures and systems of radiological significance) was the logical scope and the logical base for standardization. The reference plant concept of the March 1973 NRC policy statement was judged by GE to be an ideal approach for improving and stabilizing licensing. GE has pursued this approach since 1973 with its Nuclear Island scope and has obtained both a Preliminary Design Approval (December 1975) and Final Design Approval (July 1983). The evolution of the design from the application for Preliminary Design Approval through the present severe accident review is discussed in Section 5.

2.4 OPERATIONAL FEEDBACK

Evolution of the GESSAR II design resulted not only from new design concepts (e.g., jet pumps, internal separation) but also from analysis of operational feedback or experience. Operational feedback has been used to improve plant performance in terms of capacity factor improvements to improve overall safety. These improvements are described in Section 6.

2.5 ABNORMAL OCCURRENCES

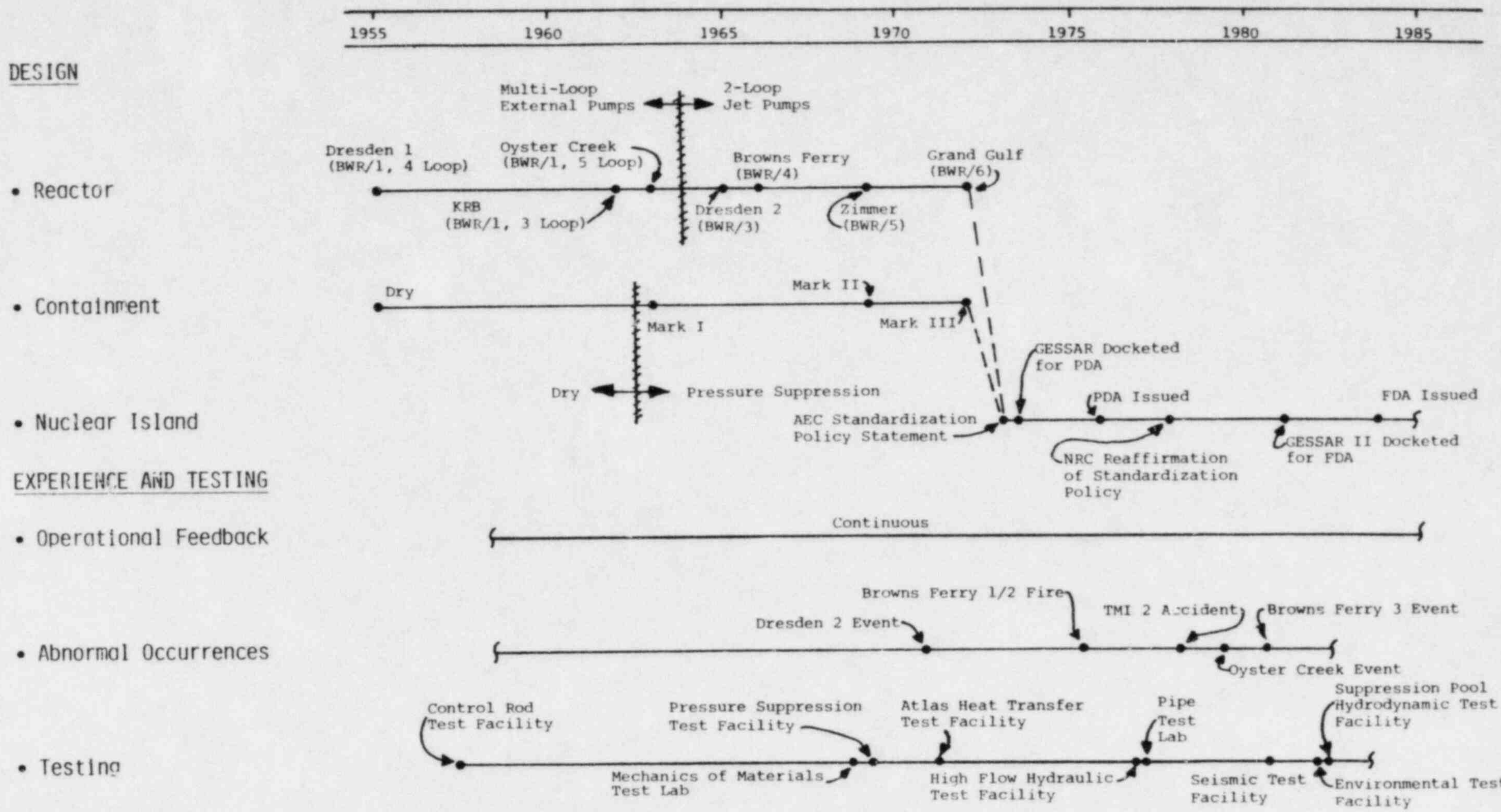
Evolution of the design has also came about through the evaluation of abnormal occurrences (e.g., Browns Ferry fire, TMI). The specific design changes and/or verification of design adequacy are enumerated in Section 7.

2.6 TESTING

Finally, disciplined design evolution has been assured through testing and the concept of "test before use." In all, more than 50 test facilities have been constructed and used to obtain design parameters and confirm design performance. Section 8 describes some of the important tests and test facilities.

FIGURE 2-1

SUMMARY OF THE EVOLUTION OF
GESSAR II BWR/6 NUCLEAR ISLAND DESIGN



3. REACTOR EVOLUTION

General Electric's path, in developing its current BWR product from early prototypes, was one of evolutionary simplification of the design. Each step in the process was taken only after thorough testing to ensure that the simplification was possible and represented an actual improvement in the design.

3.1 EARLY REACTORS

The first generation of BWRs, referred to as the "BWR/1" class, was a series of demonstration plants to evaluate various alternative features. To support the design work on the first large BWR, Dresden 1, GE built the 5-MWe VBWR. This reactor first sent steam to PG&E's turbine in 1957, proving the practicality of using a direct cycle. It was also a valuable test bed for fuel.

Dresden Unit 1 was a 210 MWe BWR introduced in 1955. This plant was started up in 1959 and was characteristic of GE's first generation of commercial BWR's. It exhibits many features similar to a pressurized water reactor, having a reactor vessel, primary and secondary loops and four steam generators (see Figure 3-1). A major difference between the Dresden 1 BWR and early PWRs is that the Dresden 1 design provided steam flow to the turbine from a dual cycle utilizing both an elevated steam drum and the steam generators.

The steam drum in Dresden 1 enabled exploration of direct-cycle operation in the early BWRs, while retaining the proven capability of the indirect cycle. In early tests, steam flow from the steam drum to the turbine was varied, relative to that from the steam generators, to demonstrate the feasibility and practicality of direct-cycle operation.

Dresden 1 followed load by controlling the temperature of the water returning to the reactor from the steam generators. Lower temperatures resulted in increased neutron moderation, and thus increased power levels.

The other eight BWR/1 plants were based upon the Dresden technology with the addition of new features for each plant. Some of the alternatives demonstrated included the following:

1. Fuel bundle arrays ranging from 6x6 to 12x12

2. A wide range of values for fuel thermal duty and core power density

3. Direct, dual and indirect cycles

4. Three different methods of steam separation, two internal and one external to the reactor

5. Two methods of recirculating the reactor water, both by natural circulation and by forced circulation

6. Various methods of load following

Consumers Power Company's 75-MWe Big Rock Point unit introduced the first BWR high power density core, 17.2 Kw/ft. This unit, which started up in 1963, demonstrated the use of flow control to change reactor power. The Big Rock Point unit was used for a 4½-year fuel R&D program during which many alternatives, including different fuel and cladding materials, higher power densities, and fuel duties, were evaluated. It was the first central-station nuclear power plant to use mechanically-sealed recirculation pumps.

PG&E's 65-MWe Humboldt Bay direct cycle unit also started up in 1963. Humboldt has no recirculation pumps, relying solely on a natural circulation core cooling. Steam separation is accomplished inside the pressure vessel, simply by gravity flow.

KRB in Germany represented GE's first major move toward BWR design simplification (Figure 3-1). In that plant, a 230 MWe unit for which design was started in 1962, the steam drum was eliminated and the steam separation and drying functions were moved inside the reactor vessel - a change which has become permanent to the BWR design. The KRB design still retained the steam generators, operating as a 3-loop, dual-cycle plant.

3.2 STEAM GENERATOR ELIMINATION AND 7X7 FUEL GEOMETRY

After the formative BWR/1 years of 1955-1962, standardizing the product was started. Besides the key design parameters, many reactor internal components were standardized, including fuel bundles, channels, control blades and drives, guide tubes, in-core neutron sensors, and steam separators.

A standard "building block" was established, consisting of a control blade, the four adjacent fuel assemblies, and the associated control rod guide tube, control rod drive and external hydraulic control unit. The quantity of blocks was varied to achieve the desired plant size.

The BWR/2 class used the best BWR/1 features. These plants used a 7x7 fuel geometry. Peak pellet exposures increased from about 30 to 42 GWd/MT. They have forced circulation pumping systems, with each pump motor being driven through a motor-generator set so the pump speed can be varied for flow control. Internal steam separators and dryers, a direct cycle, and up to five external recirculation pumping loops are used. All external steam generators were eliminated in the BWR/2 designs.

For the emergency core cooling system, a low pressure core spray system and an automatic pressure relief system were used. All of these new BWR/2 design features were incorporated into the 640 MWe Oyster Creek unit, introduced as the first BWR/2 in 1963.

3.3 JET PUMPS, IMPROVED CORE COOLING SYSTEMS AND BURNABLE POISON IN FUEL

In 1965, the use of internal jet pumps was incorporated into the design of 810 MWe Dresden 2. This change established the BWR/3 class of reactors.

Using internal jet pumps to aid recirculation flow reduced the number of external recirculation loops from five to two. The use of jet pumps eliminated the large pipes connected to the lower plenum region below the core. This enhanced ECCS capability by making the vessel easier to reflood after a postulated loss-of-coolant accident.

The use of jet pumps also increased the internal water annulus inside the reactor vessel, thus reducing the fast neutron fluence of the vessel wall. This reduced vessel irradiation has a positive effect of improving strength characteristics of the vessel material over the plant lifetime.

For the BWR/3 plants, low and high pressure flooding systems were added to the ECCS spray system to give BWR plants two emergency core cooling methods.

Also in 1965, the reactor core isolation cooling (RCIC) system was introduced as the replacement for the isolation condenser. Monticello, a BWR/3, was the first plant to employ this system. The containment evolutionary process, from dry to pressure suppression, simply lessoned the effectiveness of the emergency condenser. Important safety benefits of the RCIC system include the ability to always maintain or increase the water content of the reactor in the case of unexpected high leak rates, and also the ability to completely test the turbine pump at any time during reactor operation.

Finally, gadolinia was introduced in BWR/3's as a burnable poison to replace the poison curtains. Since its introduction as a burnable poison, gadolinia is being used to help maintain both reactor peaking and shutdown reactivity margins for spectral shift operation, longer operating cycles and higher burnup, while improving uranium utilization.

3.4 CORE POWER ENHANCEMENT AND PREFABRICATED CONTROL ROOM

In 1966 core power density and peak fuel duty were increased, a high-capacity steam separator was used, and the supporting systems were uprated accordingly. These changes were first incorporated into the Browns Ferry design, introducing the BWR/4 class of reactors. The Browns Ferry units are rated at 1100 MWe, versus the Dresden 2 rating of 810 MWe, and both have the same vessel size. The BWR/4 has a fuel duty of 18.5 Kw/ft., compared with the BWR/3 at 17.5 Kw/ft.

During the BWR/4 era, the prefabricated control room was introduced, which was called the Power Generation Control Complex, or PGCC. This was a major improvement in the design since it permitted factory assembly, rather than field installation, of major control room equipment.

3.5 EMERGENCY CORE COOLING SYSTEM UPGRADE AND INSTALLED INTERNALS

In 1969 two significant design changes were incorporated into the BWR/5. The first was an ECCS design improvement and upgrading which involved the substitution of a high pressure core spray for the high pressure core flooding system and injection of low pressure core cooling water directly into the vessel rather than into the recirculation loops. The second was an improvement to the reactor vessel recirculation flow control using special flow control valves rather than motor-generator-pump sets.

In addition, starting with BWR/5, the reactor vessel was shipped complete with installed internals, thus eliminating this field task.

3.6 FUEL PERFORMANCE ENHANCEMENT AND IMPROVED CONTROL ROOM

General Electric's most recent product is the modern BWR/6 introduced in 1972 in the design of the 1290 MWe Grand Gulf units. The BWR/6 is capable of producing 20 percent more power from the same size reactor pressure vessel as compared with the BWR/5. This was accomplished by decreasing the diameter of the fuel pellets, changing the fuel bundle design, and making other changes to the reactor vessel internals.

It was determined for 7x7 fuel geometry that strain localization due to pellet-to-clad interaction (PCI) at pellet interfaces (ridging) and pellet cracks can cause a small but statistically significant number of fuel rod perforations during normal reactor operation. The following fuel design improvements were made for 8x8 BWR/6 fuel to reduce PCI localized strain:

1. The fuel pellet length-to-diameter ratio is decreased from 2:1 to 1:1, which reduced ridging.

2. The fuel pellet is chamfered, which reduces ridging.

3. The maximum linear heat generation decreased from 18.5 Kw/ft to 13.4 Kw/ft which reduced thermal distortion and ridging.

4. The cladding heat treatment procedure is improved to reduce the variability of the cladding ductility.

In addition to the above, barrier fuel has been added to counter PCI. The 8x8 fuel geometry coupled with the zirconium barrier not only guards against PCI during load swings, but also increases fuel reliability and simplifies operations while protecting the core during inadvertent power increases.

The fuel geometry change from 7x7 to 8x8 permitted a 10 percent increase in power output. The reactor internals were redesigned to accommodate more fuel in the reactor vessels, providing for additional increase in power output. While reducing fuel duty from 18.5 to 13.4 Kw/ft, peak exposures were increased from 42 to 50 GWd/MT (early 8x8's were limited to 44 GWd/MT).

The narrower annulus around the core was made possible by a more compact and efficient multi-nozzle jet pump design, which also delivered the higher core flow required for the larger core. The increased steam flow was accommodated by an improved steam separator design.

A new compacted control room concept was introduced. Earlier control rooms had followed the typical industry practice of grouping full-sized controls by system on long rows of benchboards. For BWR/6, the operator's control console has been condensed, putting the most meaningful instruments and controls within easy reach of one person. This can be an advantage in those operating situations where operator action, if taken immediately, can avoid a forced outage. Those instruments and controls associated with automatic safety features, or with long response, were relocated to separate nearby benchboards. Further compaction of the operator console was then achieved by utilizing miniaturized instruments and controls.

The Nuclenet 1000 Control Complex is also an improved control room feature of the BWR/6. With this system, information is presented to the operator on TV screens, in the form of computer-driven color displays. The operator selects from a large assortment of available displays. The objective is to present information in a manner which makes it easiest for the operator to understand what action is needed to keep the plant safety on line.

A solid-state reactor protection system, replacing the earlier relay system, was introduced on the BWR/6. This solid-state system reduces spurious scrams and increases scram reliability.

The radioactive waste treatment system, providing for collection, processing and reclaiming of liquid waste, was developed specifically for the BWR/6. The design of this system was greatly influenced from the analysis of operating plant data.

Refueling design improvements were made on the BWR/6 to increase the capacity factor. These improvements included the use of water to promote more rapid shield removal, multiple stud tensioners for faster pressure vessel head removal and replacement, faster fuel transfer equipment, and improved mechanical reliability for the refueling platform, underwater television camera and refueling grapple.

There have also been other changes in BWR/6. For example, to aid in construction, wider use of prefabricated modules is employed. The offgas treatment system was redesigned into compact, factory-built, skid-mounted modules, and a prefabricated filter/demineralizer module for the reactor water cleanup system was introduced.

3.7 SUMMARY

A summary of BWR reactor evolution is provided in Table 3-1. Figure 3-2 shows the reactor assembly of the current product of evolution, the BWR/6 reactor.

TABLE 3-1
SUMMARY OF REACTOR EVOLUTION

FEATURE	BWR/1	BWR/2	BWR/3	BWR/4	BWR/5	BWR/6
	REACTOR TYPE					
<u>FUEL</u>						
o GEOMETRY	6x6 to 12x12	7x7	7x7	7x7	7x7	8x8
o Maximum Linear Power (KW/FT)	10.3 to 17.2	17.5	17.5	18.5	18.5	13.4
o Peak Pellet Exposure (GWd/MT)	~30	42	42	42	42	44 to 50
o Burnable Poison						
- External (Curtains)	X	X	X			
- Internal			X	X	X	X
o Barrier Fuel						X
<u>RECIRCULATION</u>						
o Natural Circulation	X					
o Multi-Loop, External Pumps	X	X				
o 2-Loop, Jet Pumps			X	X	X	X
<u>STEAM SEPARATION</u>						
o External	X					
o Internal	X	X	X	X	X	X
<u>STEAM CYCLE</u>						
o Indirect	X					
o Direct	X	X	X	X	X	X
<u>ECCS/RCIC</u>						
o Low Pressure Spray	X	X	X	X	X	X
o Pressure Relief	X	X	X	X	X	X
o High Pressure Flood			X	X	X	X
o Low Pressure Flood						
- Recirc Injection			X	X		
- Vessel Injection					X	X
o RCIC			X	X	X	X
o High Pressure Spray					X	X
<u>CONTROL ROOM</u>						
o Conventional	X	X	X			
o PGCC				X	X	X
o Solid State RPS						X
o Compacted						X
o Nuclenet						X

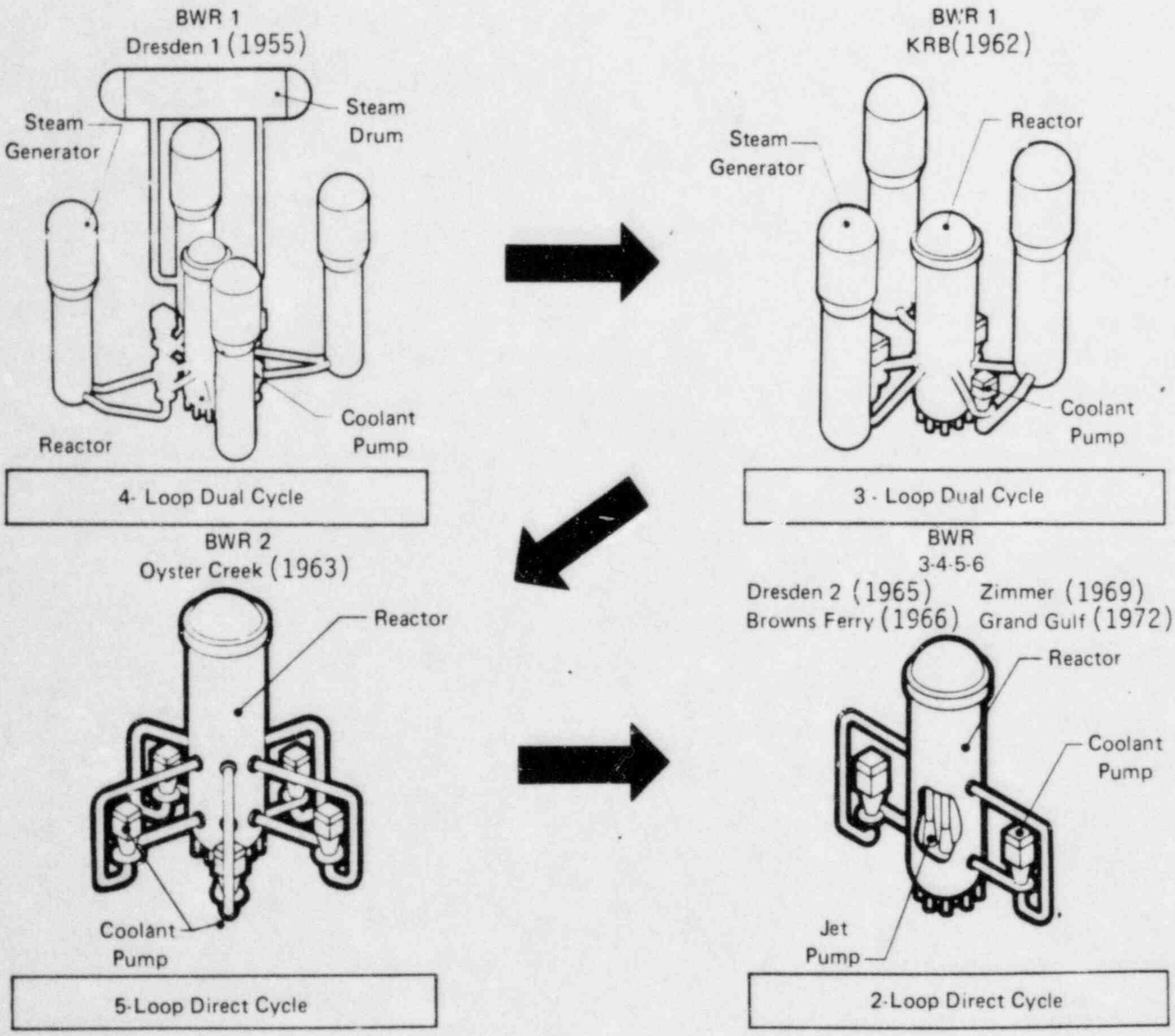


Figure 3-1. Reactor Evolution-Year of Introduction

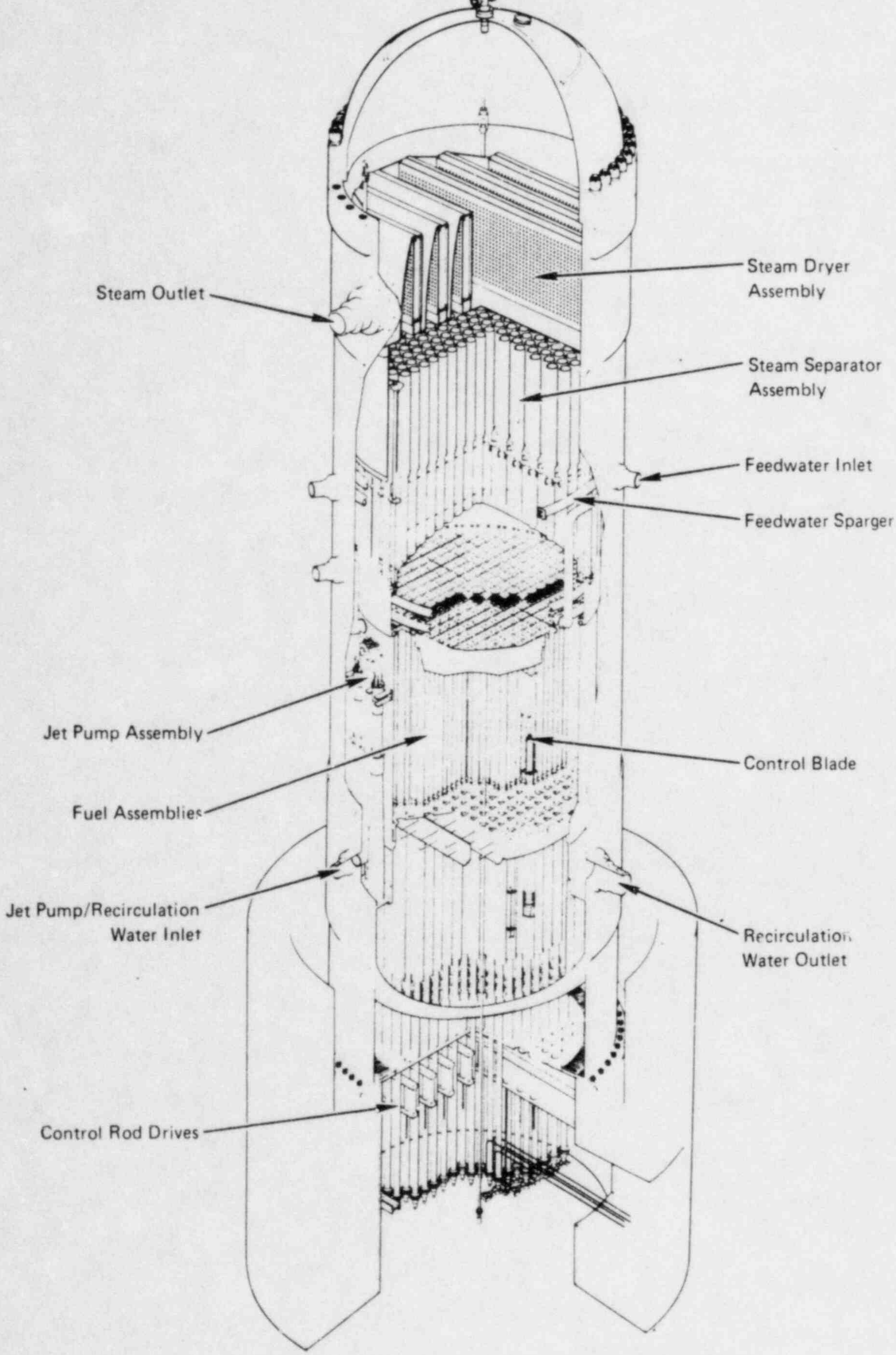


Figure 3-2. Cutaway of the BWR/6 Reactor Vessel

4. CONTAINMENT EVOLUTION

Early GE BWRs were housed in dry containments. However because of the advantages of pressure suppression (see Section 2.2), GE introduced pressure suppression containments in the early 1960's and continue to offer it today. The evolutionary relationship between the reactor and containment is shown in Table 4-1. The BWR containment evolution and design parameters is illustrated in Figure 4-1.

4.1 DRY CONTAINMENT

The purpose of a containment system is to contain any radioactive products that escape from the reactor during a postulated loss-of-coolant accident. To do this, the containment barrier must not only be leaktight, but must withstand the loadings generated by a postulated accident, including the initial pressure buildup as the escaping coolant flashes into steam. Dry containments accomplish this by providing enough volume to allow all of the escaping steam to expand without exceeding the containment vessel design pressure.

Spherical dry containments were used on some of the BWR/1 units. The 210-MWe Dresden 1 reactor is housed in a 190-foot-diameter sphere designed for about 30 psi.

4.2 PRESSURE SUPPRESSION

To reduce the size of the containment for PG&E's Humboldt Bay unit, a new design, the pressure suppression containment, was developed. It works by condensing the escaping steam in a pool of water. The reactor system is enclosed in a drywell, and large vent pipes connect the drywell to a suppression chamber. The suppression chamber has two regions: the suppression pool and an air space.

This assembly forms the leaktight containment structure. In the event of an accident, the escaping steam would at first cause some pressure rise in the drywell, but would then be exhausted underwater, causing rapid condensing and pressure relief. In addition, if fission products did escape from the fuel and were in the steam or air leaving the drywell, they would be largely retained in the pool water.

To confirm the Humboldt Bay design, test facilities, including a full-scale segment of the suppression chamber, were built at PG&E's Moss Landing Steam Plant. BWR/1 plants in India and the Netherlands later used variations of this containment design. Starting with BWR/2, BWRs used only pressure-suppression containments, while pressurized water reactors continued to use dry containments. Over the years, as unit ratings increased, PWR dry containments evolved from the steel spherical vessels to reinforced post-tensioned concrete cylindrical vessels with domed roofs, with design pressures up to 65 psi.

The BWR/2 has more coolant inventory in its primary system than a PWR of similar rating because of the larger reactor vessel that houses the steam separators and dryers. Because of the higher total coolant energy in the BWR primary system at that time, dry containments would not have been as economical for BWRs as for PWRs. Offsetting this, BWRs have a lower rate of coolant loss because of the lower system pressure and smaller pipe sizes. This lower rate of energy release lends itself well to the use of pressure suppression containment.

4.2.1 Mark I

The pressure suppression configuration that became the first standard pressure-suppression containment for BWRs is called the "Mark I" design. In the Mark I design, the lightbulb-shaped drywell and the doughnut-shaped suppression chamber are joined by large ducts connected to a header. Individual downcomers channel the steam from the header into the suppression pool.

The Mark I containment is used for BWR/2, BWR/3 and most BWR/4 units. This drywell shape resulted from the need for a small size removable closure at the top for refueling and a wide space at the bottom to enclose the reactor recirculation pumps. The typical Mark I containment boundary is a steel vessel designed for 62 psi.

4.2.2 Mark II

The next containment evolution was the "over-under" or Mark II configuration. In Mark II, the suppression chamber is located under a drywell that has the shape of a truncated cone. These two regions are connected by straight pipes. The Mark II containment is used for a few of the later BWR/4 units and for BWR/5 units.

Mark II differs from Mark I in that it has more room in the drywell, particularly in the region of the steam piping and ECCS piping in the upper region and the recirculation loops in the lower region.

This configuration resulted in a lower design pressure of 45 psi. It also permitted a wider variety of construction materials for the drywell and suppression chamber, such as post-tensioned or reinforced concrete with a steel liner, as well as freestanding steel vessels. Overall, Mark II resulted in a smaller reactor building with better utilization of space.

4.2.3 Mark III

The Mark III containment is designed for use with the BWR/6 system. Mark III combines the proven safety and low pressure advantages of a pressure-suppression containment with a simple cylindrical shape. The drywell, enclosed by the suppression-chamber air space, is completely separated from the containment barrier.

The containment is enclosed by a reinforced-concrete shield building that protects it against external missiles and provides post-accident radiation protection.

Together, the Mark III reactor building, fuel building and auxiliary building perform the same functions as the reactor building that enclosed the Mark I or Mark II containment. The auxiliary and fuel buildings along with the shield building serve as the secondary containment.

Some differences of the Mark III design include the following:

1. The reactor pressure vessel pedestal is shorter, improving the seismic response of reactor vessel and its internals.

2. Enclosing all primary piping inside the reinforced concrete drywell or steam-tunnel walls minimizes the possibility of damage to the containment barrier by pipe failure or by jet impingement of escaping coolant.

3. The design pressure is reduced to 15 psi, which translates to a free-standing steel containment with a thickness of 1-3/4 inches or less, eliminating the need for stress relieving during field construction.

4. The cylindrical drywell shape is easy to construct, and it has more room for equipment installation and in-service inspection.

5. More construction sequences are available, providing a variety of critical path alternatives to accommodate inevitable construction or equipment delivery delays.

6. A choice of construction techniques can be used, such as slip-formed shield building, prefabricated pedestal and reactor shield wall, and larger prefabricated piping and equipment modules.

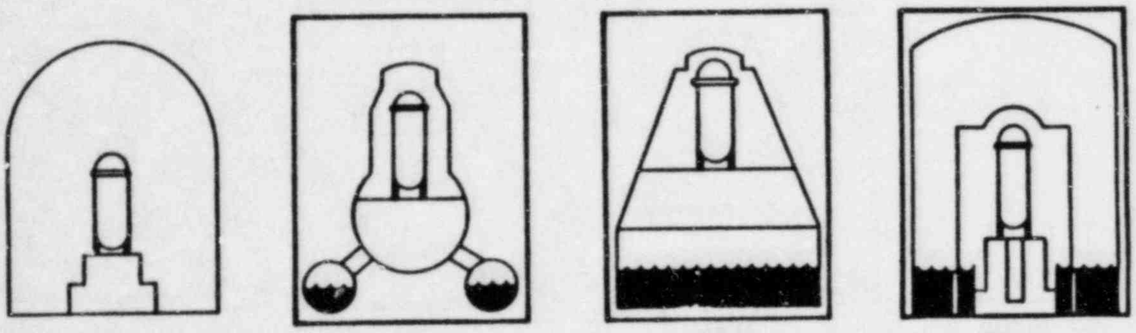
Several Mark III containments have already been placed into operation - the 950-MWe Kuosheng Unit 1 in Taiwan in 1981, and most recently the 1290-MWe Grand Gulf Unit 1 operated by Mississippi Power and Light Co.

The Mark III design contributes to faster refueling times. In order to provide radiation protection above the reactor during operation, shielding is necessary. On Mark III, this shielding is provided by a water-filled pool over the drywell head, instead of concrete shield blocks as on Mark I and Mark II plants. This water can be pumped out for drywell head removal in 2 hours, compared to about 8 hours consumed on the average in removing the shield blocks.

TABLE 4-1

SUMMARY OF CONTAINMENT EVOLUTION

CONTAINMENT TYPE	REACTOR TYPE					
	BWR/1	BWR/2	BWR/3	BWR/4	BWR/5	BWR/6
DRY	X					
PRESSURE SUPPRESSION						
o Developmental	X					
o Mark I		X	X	X		
o Mark II					X	
o Mark III						X



	DRY	MK I	MK II	MK III
PRESSURE SUPPRESSION	NO	YES	YES	YES
NUMBER OF BARRIERS	1	2	2	3
CONTAINMENT VOLUME MILLIONS CU FT	2.5	0.4	0.5	1.6
HEAT CAPACITY BILLIONS BTU	0.3	1.7	1.3	1.3
LOCA PRESSURE PSIG	50	44	42	9
DESIGN PRESSURE PSIG	50	62	45	15

Figure 4-1. Containment Evolution

5. NUCLEAR ISLAND EVOLUTION

5.1 INITIAL STAGE

In late 1971, when the BWR/6 reactor and Mark III containment were at the early stages of design, GE knew it had a new design offering with significantly improved safety margins (e.g., ECCS, etc.). However, GE certainly was not looking forward to repetitive reviews and the "maybe better, but not licensed" perspective. So GE met with AEC* and the full ACRS for some very candid discussions. In essence GE indicated that it would appreciate a safety review of BWR/6 and Mark III containment even before the first project review of this BWR version. Hence, the first step toward Nuclear Island evolution began with the filing of topical reports in June 1972 describing the BWR/6 reactor and Mark III containment. As a result of this early submittal and numerous meetings with the AEC/ACRS, GE received letters from the ACRS in November of 1972 on BWR/6 and January of 1973 on Mark III containment that were quite favorable.

5.2 SELECTION OF SCOPE

The initial NRC policy statement on standardization of nuclear power plants was issued in April 1972. It provided the impetus for both industry and the NRC to initiate active planning in their respective areas in order to realize the benefits of standardization while maintaining protection for the health and safety of the public and the environment. In a subsequent statement issued March 1973, the NRC announced its intent to implement a standardization policy for nuclear power plants. THe reference system concept of the 1973 standardization program provides for an applicant to submit a standard safety analysis report (SSAR) describing the system or nuclear power plant features in

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* Currently the NRC

order to obtain a preliminary design approval on either. The preapproved system or plant design then could be referenced by a utility applicant in a preliminary safety analysis report at the construction permit stage, thus eliminating the need for a custom review of the design for each applicant.

The reference plant concept was judged by GE to be an ideal approach for improving and stabilizing licensing, and GE has pursued this approach vigorously since 1973.

With the issuance of these policy statements, it was necessary to ask some key questions about the traditional GE scope of supply:

1. How effective can an AEC signoff of only the NSSS be? It is not too effective if only 20% of the SAR technology can be standardized.

2. If NSSS only were described, how does the complication of the interface with the containment, the ECCS, the power considerations, the seismic criteria, etc. reduce the effectiveness of a standard plant application?

It just did not look like an NSSS scope was the appropriate approach. Therefore, GE concluded that the Nuclear Island, i.e., all parts of the plant that have radiological significance, was the logical scope of the General Electric Standard Safety Analysis Report (GESSAR)* and the logical base for standardization. GESSAR contains safety information for a BWR/6 Mark III nuclear power plant, including the nuclear steam supply system (NSSS), the engineered safety feature systems, the containment and auxiliary buildings, the control room, radioactive waste system and related systems and structures.

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* Preliminary Design Approval version of GESSAR II

Not included in the GESSAR Nuclear Island scope are the turbine-generator and auxiliaries, the turbine building, portions of the main steam system (beyond the main steam shutoff valves located in the auxiliary building), the main condenser, the circulating water system and intake structure, condensate storage facilities, offsite electrical power, and the ultimate heat sink, raw and potable water systems, parts of the service and instrument air systems outside the nuclear island, the auxiliary steam system and, of course, the site.

5.3 PRELIMINARY DESIGN APPROVAL

On April 30, 1973, GE filed GESSAR for a Preliminary Design Approval (PDA). On July 30, 1973 the NRC docketed the GESSAR application. The review of GESSAR was carried out by the NRC pursuant to Appendix O to 10CFR Part 50, "Licensing of Production and Utilization Facilities," using a procedural sequence similar to that used for custom plant reviews. The initial phases, that is, preliminary review, question rounds, etc., were analogous to the normal construction permit stages of review.

The technical review of GESSAR by the NRC included:

1. The site design envelope parameters including the wind loadings, design bases tornado, design bases flood evaluation, the design bases earthquake, the snow loading, and maximum precipitation.

2. The design and expected performance of the Nuclear Island's structures, systems, and components important to safety to determine whether they are in accord with the Commission's General Design Criteria (GDC), the Commission's Quality Assurance Criteria, and other applicable guides, codes and standards, and whether any departures from criteria, codes and standards were identified and justified.

3. The expected response of the facility to various anticipated operating transients and to a broad spectrum of postulated accidents to determine that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered. The NRC performed conservative analyses of these design basis accidents and determined that the calculated potential offsite doses that might result in the very unlikely event of their occurrence would be well within the Commission's guidelines for site acceptability, as given in 10CRF Part 100, for typical sites.

4. The design of the systems provided for control of the radiological effluents from the plant to determine that these systems, in conjunction with an acceptable BOP design, reasonable meteorology and site boundaries, are able to control the release of radioactive wastes within the limits of the Commission's regulations in 10CRF Part 20 and that the plant will be operated in such a manner as to reduce radioactive releases to levels that are as low as practicable in accordance with the Commission's regulations in 10CFR Part 50.

5. New control rod position detection system; a new method of increasing the negative reactivity during a scram to cope with changes to scram-reactivity during core life; the use of ganged control rods; a revised rod pattern control system; and a solid state, 2-out-of-4 protection system.

6. GE Thermal Analysis Basis (GETAB) based on further boiling transition tests.

7. Post-LOCA H_2 generation and control, a main steam line sealing system, suppression pool bypass and testing, drywell structural and leakage testing, quality classification of main steam radwaste and auxiliary systems and interface definition and quantification.

As a consequence of the NRC review, a number of changes were made to the GESSAR design. Examples of these changes are summarized in Table 5-1.

Following more than 1100 questions and 11 man-years of exhaustive review by the NRC and its subcontractors, and a favorable review by the ACRS, the NRC issued PDA-1 for the GESSAR Nuclear Design on December 22, 1975.

5.4 FINAL DESIGN PERIOD

Between issuance of the GESSAR PDA in December 1975 and the submittal of GESSAR II for an FDA in March 1980, GE continued with the final design, GESSAR, II by utilizing the TVA Hartsville and Phipps Bend facilities. Both of these facilities received construction permits, Hartsville in May 1977 and Phipps Bend in January 1978, referencing the GESSAR FDA-1.

All of the significant changes made in the Nuclear Island design between issuance of the PDA and submittal for an FDA are listed in Table 5-2.

5.5 FINAL DESIGN APPROVAL AND SEVERE ACCIDENT REVIEWS

On March 31, 1980, GE filed GESSAR II for a Final Design Approval (FDA). On December 9, 1981, the NRC docketed the GESSAR II application. NRC technical review and evaluation of GESSAR II considered the principal matters summarized below:

1. The design, fabrication, and testing criteria, and expected performance characteristics of the system and components important to safety to determine that they are in accord with the Commission's General Design Criteria (GDC), Quality Assurance Criteria (QAC), Regulatory Guides, and other appropriate rules, codes and standards, and that any departures from these criteria, codes and standards, have been identified and justified. Although

GESSAR II was only required to meet the regulatory guidelines in effect at the time of docketing the staff has reviewed the GESSAR II application to the Standard Review Plan (SRP) (NUREG-0800) in accordance with 10CFR50.34(g).

2. The expected response of GESSAR II to various anticipated operating transients and to a broad spectrum of postulated accidents. Based on this evaluation, NRC determined that the potential consequences of a few highly unlikely postulated accidents (design-basis accidents, DBAs) would exceed those of all other accidents considered. The NRC performed conservative analyses of these DBAs to determine that the calculated potential offsite radiation doses that might result, in the very unlikely event of their occurrence, would not exceed the Commission's guidelines for site acceptability given in 10CFR 100 for the GESSAR II site envelope.

The key changes made to the GESSAR II design during the NRC FDA and severe accident reviews are provided below:

1. Scram discharge volume. A number of changes were made in the scram discharge volume by adding redundant and diverse instrumentation to provide reliable scram. Also added redundant vent and drain valves to satisfy the single failure criterion.

2. Containment strength. The dome portion of the containment was redesigned to satisfy the load requirements of 45 psig Service Level C [10CFR50.34 (f) - CP/ML Rule].

3. Control room and human factors. The GESSAR II design contains a solid state control room with the important controls and instrumentation easily accessible by the operators. This design has been reviewed by the NRC for human factors. In addition, the GESSAR II control room has been upgraded to include the Emergency Response Information System which provides the Safety Parameter Display capability for both plant maintenance and operation.

4. TMI changes. In addition to the above, the GESSAR II design incorporates all of the NUREG-0737 improvements required for BWRs.

5. Anticipated Transients Without Scram (ATWS). The GESSAR II design is committed to incorporate the requirements resulting from the current rulemaking on ATWS. These requirements include alternate rod insertion, recirculation pump trip and an enhanced standby liquid control system. Incorporation of these features provides a significant reduction in risk associated with failure to scram.

6. Ultimate Plant Protection System (UPPS). This system is to be used during extended station blackout. It is composed of diesel driven fire pumps, fire truck or other pumping capability outside the containment linked with a system of piping which will remotely depressurize the reactor and permit core cooling for an indefinite period of time. No conventional controls, AC power, DC power, or other systems are required for this operation.

   The containment vent is an integral part of UPPS. In the event of the loss of RHR or other means of heat removal, the suppression pool can be utilized to store large quantities of heat. In order to enhance the effectiveness, the containment can be vented by the operator using Emergency Procedure Guidelines. Furthermore, for postulated severe accidents given the filtering capability of the pool there should be no concern for exposing the public to radiation hazard. This containment venting capability provides both heat removal and means for protecting the containment from overpressure.

The above changes and others are summarized in Table 5-3.

Following another exhaustive review by the NRC and its contractors (nearly 600 question and an excess of 12 man-years) and a favorable review by the ACRS, the NRC issued FDA-1 for the GESSAR II design on July 27, 1983. Unfortunately, this FDA was limited for incorporation by reference in applications for operating licenses for those plants that referenced PDA-1 at the construction permit stage (i.e., only Hartsville). The NRC found this restriction necessary because of the lack of a Severe Accident Policy.

approval of 10

The NRC's review of the GESSAR II severe accident design (another 300 questions and 6 man-years) is nearing completion with ACRS meeting to begin in August 1984. Hopefully, the Severe Accident Policy will be passed shortly so that the FDA-1 restrictions can be removed.

TABLE 5-1

GESSAR CHANGES DURING
PRELIMINARY DESIGN APPROVAL REVIEW

- An increase in the wind loading, snow and ice loadings and elevation of ground water with respect to the foundation mat to permit the plant to be used on more sites.

- The seismic instrumentation program was augmented.

- A mainsteam line leakage control system was added.

- The RCIC system was upgraded to an engineered safety feature.

- The operability of active components will be verified by testing.

- Design measures were taken to protect against the dynamic effects associated with pipe breaks.

- A finite element methodology is used to analyze various soil conditions to evaluate soil structure interactions.

- Fuel building was upgraded to withstand tornado missiles.

- Methods for Seismic Analysis compliance with requirements of Regulatory Guide 1.60 and 1.61.

- Main Steam Line and Feedwater Piping Reclassification.

- Mark III containment changes to accommodate pool swell testing results.

- Increased drywell design pressure margin.

TABLE 5-1
(Continued)

GESSAR CHANGES DURING
PRELIMINARY DESIGN APPROVAL REVIEW

- Tests to verify that controls on stainless steel are adequate to prevent sensitization.
- GE agreed to preoperational vibration tests on Class 1 and 2 piping systems.
- Reduced containment leakage design criteria.

TABLE 5-2

SIGNIFICANT CHANGES BETWEEN ISSUANCE OF
PRELIMINARY DESIGN APPROVAL AND APPLICATION FOR FINAL DESIGN APPROVAL

- Added positive leakage control systems to supply sealing medium (air/water) to pressurize space between isolation valves of select lines thereby preventing bypass leakage following postulated LOCA.

- Increased allowable primary containment leakage rate from 0.3%/day to 1.0% day since positive leakage control systems reduce bypass leakage thus enabling higher filtered release rates.

- Incorporated stainless steel cladding of the carbon steel containment vessel in the wetted areas of the suppression pool to protect against pitting and corrosion, reduce pool maintenance, operating costs over the life of the plant, and crud accumulation.

- Added suppression pool cleanup system to improve reliability of plant operations.

- Incorporated state-of-the-art buckling methodology for stability analysis of the containment vessel.

- Added motor-generator sets to provide control for reduced flow during startup and shutdown to improve operation.

TABLE 5-2
(Continued)

- Deleted recirc pump/motor decoupler from design since analysis demonstrated that there are no unacceptable consequences for postulated LOCA event.

- Added low-low set relief logic to assure that no more than one safety/relief valve cycles subsequent to the first pressure peak. This maintains design basis for containment loads and overpressure transients.

- Reduced magnitude of SRV load definition by approximately 35% to reflect recent in-plant test data.

- To improve fuel performance, the number of water rods in each fuel bundle has been changed from 1 to 2. Five different U-235 enrichments are now used in the fuel.

- The containment cylindrical shell is backed by structural concrete below elevation (-) 5 ft., 3 in., in the annular space to make the lower portion of the containment more rigid and thus reduce the structural response due to SRV loadings.

TABLE 5-2
(Continued)

- CRD return line to RPV deleted to reduce nozzle cracking problem.

- Changed feedwater sparger thermal sleeve to provide improved slip fit design of sparger to nozzle to eliminate failure, leakage, and provide for possible in-service inspection.

- Added more fuel storage castings in racks for use in spent fuel, and containment pool areas to increase capacity to handle more onsite fuel storage.

TABLE 5-3

GESSAR II CHANGES DURING
FINAL DESIGN APPROVAL AND SEVERE ACCIDENT REVIEWS

- Scram discharge volume fully instrumented and redundant volumes substantially reduce risk of partial scram.

- Added remote shutdown station divisionally separated with essential equipment qualified as safety-related.

- Material changeout of reactor coolant boundary material to avoid intergranular stress corrosion cracking.

- Added all NUREG-0737 TMI improvements required for BWRs.

- Increased containment vessel structural capability to provide additional design margin.

- ESF filter temperature and pressure instrumentation added in accordance with SRP 6.5.1.
- Increased detergent waste tank from 1500 to 10,000 gallons.

- Eliminated Division 1 and 2 AC and DC cross-ties.

- Utilization of 3-hour rated ductwork in-lieu of fire dampers in safety grade single-duct systems.

- Demonstrated that soil-structure interaction analysis envelopes grade level input motion and elastic half-space model.

TABLE 5-3

(Continued)

GESSAR II CHANGES DURING

FINAL DESIGN APPROVAL AND SEVERE ACCIDENT REVIEWS

- Increased factor of safety against containment buckling to further insure containment integrity against localized loads.

- Changed concrete structures code requirements from ACI-318 to ACI-349.

- Added Ultimate Plant Protection System (UPPS) for use during extended station blackout. UPPS uses no conventional controls, AC power or DC power to provide reactor depressurization, reactor makeup and heat removal.

- Upgraded the control room to include the Emergency Response Information System to provide the safety parameter display capability for both plant maintenance and operation.

- Added alternate rod insertion, recirculation pump trip and an enhanced liquid control system to provide a significant reduction in risk associated with failure to scram.

TABLE 5-3
(Continued)

- Long term containment repressurization reduced from 90 to 50% of design value.

- Added separate containment purge lines and reduced diameter from 18 to 9 inches.

- Modified design for full compliance with post accident monitoring requirements (Reg. Guide 1.97, Rev. 2)

- Changed from 50 to 3 micron filters and from carbon to galvanized pipe in instrument air and pheumatic supply systems to increase reliability of these systems.

- Re-analyzed reactor building for concrete annulus and increased response spectra to envelope results.

- Added smoke detectors at selected locations to meet the requirements of SRP 9.5.1.

- Added dual (series) valve barriers for selected test, vent and draw connections to meet containment isolation requirements.

6. EVOLUTION THROUGH ANALYSIS OF OPERATIONAL FEEDBACK

Analysis of feedback from operating reactors is an integral part of the evolutionary process. In fact, many of the design changes described in Sections 3, 4 and 5 were improvements identified by the operational feedback analysis process. This section briefly describes the process itself and provides examples of design changes that have resulted from this feedback analysis.

6.1 OPERATIONAL FEEDBACK ANALYSIS PROCESS

Within the GE Engineering organization an independent group called the Reliability Engineering Operation, analyzes feedback from operating plant gathered by field service departments. In-plant data from around the work is fed into the computer daily. It is extracted for study by the reliability engineers who prepare periodic trend reports showing BWR performance by system and major component. The line organization and the reliability engineers review these reports carefully to define recommended improvements for incorporation into the design of new plants and for potential backfit into existing plants. Such improvements enhance plant safety directly or indirectly through improved plant performance (i.e., increased capacity factor which translates into improved safety).

6.2 DESIGN CHANGES

Since operational feedback analysis is an integral part of the evolutionary process, essentially all design changes identified by the analysis of operational data have been identified in the design evolution sections (Sections 4, 5 and 6). Hence, the presentation here will be limited to identifying the design changes identified in the design evolution sections that are attributable to the analysis of operational feedback. The specific design changes are provided below:

1. 7x7 fuel geometry (Section 3.2)*

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* ( ) is the design evolution section describing the change.

2. 8x8 fuel geometry (Section 3.6)

3. Increased fuel pellet exposure (Sections 3.2 and 3.6)

4. Burnable poison in fuel (Section 3.3)

5. Barrier fuel design (Section 3.6)

6. Power Generation Control Complex (Section 3.4)

7. Solid state reactor trip system (Section 3.6)

8. Compacted Control Room (Section 3.6)

9. Nuclenet (Section 3.6)

10. Radioactive Waste Treatment System (Section 3.6)

11. Refueling design improvements (Section 3.6 and Subsection 4.2.3)

12. Feedwater sparger design improvement (Table 5-2)

13. Elimination of CRD return line (Table 5-2)

14. Suppression pool stainless steel liner (Table 5-2)

15. Addition of suppression pool cleanup system (Table 5-2)

16. Number of water rods/bundle increased from 1 to 2 (Table 5-2)

17. Added Ultimate Plant Protection System (Table 5-3)

18. Upgraded control room to include ERIS (Table 5-3)

7. EVOLUTION THROUGH EVALUATION OF ABNORMAL OCCURRENCES

In addition to evolution through the analysis of operational feedback described in Section 6, is evolution through the evaluation of abnormal occurrences. This section addresses five of the more noteworthy abnormal occurrences and how their evaluation have influenced the GESSAR II design. These abnormal occurrences are presented in chronological order.

7.1 DRESDEN 2 EVENT

7.1.1 Description

The Dresden 2 event occurred in June 1970. The unit was operating at normal water level with the safety valves closed, the main steam isolation valves open, and the feedwater system operating normally. A transient was initiated when the main turbine tripped, initiating a reactor scram.

Feedwater control problems occurred and the reactor pressure in the steam line reduced to about 850 psi. About a half a minute after the initiation of the transient, the containment and main steam lines isolated. During this time period, feedwater was continuing to flow into the reactor, and the operator, at that point, assumed manual feedwater control. Unfortunately, he relied on a faulty strip-chart recorder (which had a stuck pen) for reactor water level indication. This recorder indicated that liquid level was at a normal or low value, so the operator held the feedwater flow on past the point at which it actually was required to keep the vessel at normal level. Eventually, the water overflowed into the steam lines, opening one safety valve. The jet of steam and water from this open safety valve impinged on the lifting levers of two adjacent valves, causing them to open slightly and keeping them slightly open. The open safety valves discharged steam and water into the drywell, damaging cables and equipment. The unit was shut down for about two months before returning to power.

7.1.2 Lessons Learned

As a result of this incident, an evaluation was made of the BWR and its response to such initiating events, and the BWR design was modified. These modifications were the "Lessons Learned" from the Dresden 2 event and included the following:

1. The operator should not rely on a single level indication because it can be misleading.

2. Automatic protection should be provided to prevent overfilling of the vessel.

3. Containment environment can be more severe than the then-current design basis.

4. Removal of test handles from valves would reduce the likelihood of their being opened by jet impingement from the discharge of adjacent valves.

5. Safety and relief valves should be piped to the suppression pool.

7.1.3 Actions Taken

Actions taken at GE include:

1. In the early 1970's, the Morris training facility went into operation. The syllabus at this facility has been modified from time to time to incorporate lessons learned from plant operating experience. (The facility also is used to develop new course material; it performs similar to an actual power plant and observations of its response to various initiating events often suggest the best course of operator action to be taken.)

2. Water level instrumentation and controls were modified by adding high level trips to the high pressure systems to assure that the reactor pressure vessel does not overfill.

3. The containment equipment environmental specification was upgraded to make prevision for higher temperature environment.

4. Removed test handles from valves.

5. Piping all safety and/or relief valves to the suppression pool to avoid discharging steam/water mixtures in the drywell during plant transients.

7.2 BROWNS FERRY 1/2 FIRE

7.2.1 Description

A fire was experienced at the Browns Ferry Plant in March 1975.

Units 1 and 2 share a common room with a cable spreading room located beneath the control room. Cables carrying electrical signals between the control room and various pieces of equipment in the plant pass through the cable spreading room.

The immediate cause of the fire was the ignition of polyurethane foam which was being used to seal air leaks in cable penetrations between the Unit 1 reactor building and a cable spreading room located beneath the control room of Units 1 and 2. The material ignited when a candle flame, which was being used to test the penetration for leakage, was drawn into the foam by air flow through the leaking penetration.

Following ignition of the polyurethane foam, the fire propagated through the penetration in the wall between the cable spreading room and the Unit 1 reactor building. In the cable spreading room, the extent of burning was limited and the fire was controlled by a combination of the installed carbon dioxide extinguishing system and manual fire fighting efforts. Damage to the cables in this area was limited to about 5 feet next to the penetration where the fire started. The major damage occurred in the Unit 1 reactor building adjacent to the cable spreading room, in an area roughly 40 feet by 20 feet, where there is a high concentration of electrical cables. About 1600 cables were damaged. There was very little other equipment in the fire area, and the only damage, other than that to cables, trays, and conduits, was the melting of a soldered joint on an air line and some spalling of concrete.

The electrical cables, after insulation had been burned off, shorted together and grounded to their supporting trays or to the conduits, with the result that control power was lost for much of the installed equipment such as valves, pumps, and blowers. Sufficient equipment remained operational throughout the event to shutdown the reactors and maintain the reactor cores in a cooled and safe condition, even though all of the emergency core cooling systems for Unit 1 were rendered inoperable, and portions of the Unit 2 systems were likewise affected. No release of radioactive material above the levels associated with normal plant operation resulted from the event.

In addition to the cable damage, the burning insulation created a dense soot which was deposited throughout the Unit 1 reactor building and in some small areas in the Unit 2 reactor building. The estimated 4,000 pounds of polyvinyl chloride insulated cable which burned also released an estimated 1,400 pounds of chloride to the reactor building.

7.2.2 Lessons Learned

The major lessons learned from the Browns Ferry 1/2 fire investigations (Reference 1 and 2) are:

1. Nuclear power plants should use the concept of defense-in-depth to achieve the required high degree of safety by using echelons of

safety systems. This concept is also applicable to fire safety in nuclear power plants. With respect to the fire protection program, the defense-in-depth principle should be aimed at achieving an adequate balance in:

a. Preventing fires from starting;

b. Detecting fires quickly, suppressing those fires that occur, putting them out quickly, and limiting their damage; and

c. Designing plant safety systems so that a fire that starts in spite of the fire prevention program and burns for a considerable time in spite of fire protection activities will not prevent essential plant safety functions from being performed.

2. Water will promptly extinguish electrical cable fires. Since prompt extinguishing of the fire is vital to reactor safety, fire and water damage to safety systems is reduced by the more efficient application of water from fixed systems spraying directly on the fire rather than by manual application with fire hoses. Appropriate firefighting procedures and fire training should provide the techniques, equipment, and skills for the use of water in fighting electrical cable fires in nuclear plants, particularly in areas containing a high concentration of electric cables with plastic insulation.

3. Separate fire areas for each division of safety-related systems will reduce the possibility of fire-related damage to redundant safety-related equipment. Fire areas should be established to separate redundant safety divisions and isolate safety-related systems from fire hazards in nonsafety-related areas. Particular design attention to the use of separate isolated fire areas for redundant cables will help to avoid loss of redundant safety-related cables. Separate fire areas should also be employed

to limit the spread of fires between components that are major fire hazards within a safety division. Where redundant systems cannot be separated by fire barriers, as in containment and the control room, it is necessary to employ other measures to prevent a fire from causing the loss of function of safety-related systems.

Within fire areas containing components of a safety-related system, special attention should be given to detecting and suppressing fires that may adversely affect the system. Measures that can be taken to reduce the effects of a postulated fire in a given fire area include limiting the amount of combustible materials, installing fire-resistant construction, providing fire rated barriers for cable trays, installing fire detection systems and fixed fire suppression systems, or providing other protection suitable to the installation. The fire hazard analysis will be the mechanism to determine that fire areas have been properly selected.

Suitable design of the ventilation systems can limit the consequences of a fire by preventing the spread of the products of combustion to other fire areas. It is important that means be provided to ventilate, exhaust, or isolate the fire area as required and that consideration be given to the consequences of failure of ventilation systems due to fire causing loss of control for ventilation, exhausting, or isolating a given fire area. The capability to ventilate, exhaust, or isolate is particularly important to reduce the habitability of rooms or spaces that must be attended in an emergency. In the design, provision should be made for personnel access to and escape routes from each fire area.

7.2.3 Actions Taken

The design of the GESSAR II Fire Protection Systems and the Fire Hazards Analysis (GESSAR II, Appendix 9A) were in process at the time of the Browns Ferry 1/2 fire. Consequently, the GESSAR II design incorporated all of the lessons learned. This is evidenced by the conclusion in the NRC GESSAR II SER (Reference 3) that the GESSAR II fire protection features conform to the guidelines of Reference 2.

7.2.4 References

1. U.S. Nuclear Regulatory Commission, "Recommendations Related to Browns Ferry Fire," NUREG-0050, February 1976.

2. U.S. Nuclear Regulatory Commission, "Standard Review Plan," NUREG-0800, Section 9.5.1.

3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Final Design Approval of the GESSAR II BWR/6 Nuclear Island Design," NUREG-0979, April 1983.

7.3 TMI 2 ACCIDENT

7.3.1 Description

The TMI accident occurred in March 1979. This was a "small break" loss of coolant accident (LOCA) which was not recognized by the operators. The accident began with a loss of feedwater and was followed by a failure-to-reclose of the pressurizer relief valve. Because of a valve misalignment, the auxiliary feedwater system initially failed to provide water to the steam generators. The diaphragm on the reactor coolant drain tank ruptured. During the accident ECCS and main coolant flow were interrupted by the operators, primary system water was released in large quantities to the containment and auxiliary building, and fission products were released from the containment and auxiliary building. Some coolant flashed into steam in the primary loop, and some steam and/or noncondensable gases were generated in the primary loop while the core was partially uncovered. Extensive fuel damage occurred.

7.3.2 Lessons Learned

The investigations and studies associated with the TMI-2 accident produced several documents specifying results and recommendations, which prompted the issuance by the NRC of various bulletins, letters, and NUREG's providing guidance and requiring specific actions by the nuclear power industry. In May 1980, the issuance of NUREG-0660 (Reference 1) providing a comprehensive and integrated plan and listing of requirements to correct or improve the regulation and operation of nuclear facilities based on the experience from the accident at TMI-2 and the studies and investigations of the accident. NUREG-0737 (Reference 2), issued in November 1980, listed items from NUREG-0660 approved by the NRC for implementation, and included additional information concerning schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions.

7.3.3 Actions Taken

General Electric thoroughly reevaluated its BWR/6 Mark III safety features since the TMI-2 accident. Results of the reanalyses confirmed that the GESSAR II design is highly resistant to plant damage or significant off-site radiological releases resulting from not only "TMI-type" events, but also from a broad spectrum of degraded events ranging from transient events, with no-pipe-break, to large-pipe-break accidents compounded by operator error. The features which provide this protection are:

1. Thirteen high- and low-pressure pumps which provide makeup directly to the reactor vessel.

2. Rapid primary system depressurization capability which can be used to make both high- and low-pressure pumps available to maintain reactor water level for any potential accident scenario,

3. Strong natural circulation, internal to the reactor vessel, which provides passive core cooling as long as vessel water inventory is maintained,

4. Two diverse, top-entry spray systems which provide core cooling, even if reactor vessel water inventory is depleted,

5. Redundant reactor water level measurement directly on the reactor vessel to provide a reliable basis for automatic and manual initiation of plant protection systems,

6. Operation in the boiling mode, familiar to plant operators, for both normal and emergency operating conditions,

7. A common operator response, based on symptoms rather than events diagnosis, to all reactor water inventory threatening events,

8. Capability to vent noncondensible gases from the reactor vessel if necessary,

9. A large suppression pool heat sink inside the containment, which can accept decay heat, unattended, for up to 6 hours with the reactor vessel isolated from the main condenser,

10. Suppression pool "scrubbing" of fission products from safety/relief valve and loss-of-coolant accident discharges from the primary system, and

11. Secondary containment, with leakage filtration, to provide an additional barrier against potential offsite radiological release.

The results of post-TMI studies have confirmed and reinforced GE's confidence in the inherent and engineered plant protection features of the GESSAR II design. Prior to the TMI-2 accident, the GESSAR II design already incorporated many features which respond to the longer-term design trends which emerged in the aftermath of TMI-2. Finally, the GESSAR II now incorporates all of the NUREG-0737 improvements required for the BWR.

7.3.4 References

1. U.S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," USNRC Report NUREG-0660, Vols. 1 and 2, May, 1980.

2. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," USNRC Report NUREG-0737, November, 1980.

7.4 OYSTER CREEK EVENT

7.4.1 Description

The Oyster Creek event occurred in May 1979. It was initiated by a loss of feedwater flow and subsequent MSIV closure with isolation condenser operation. Oyster Creek has an isolation condenser which provides a coolant flow and heat sink when the reactor is isolated, instead of a reactor core isolation cooling (RCIC) system. It was proper for the isolation condenser to be in operation. However, the operator took an improper action and manually closed all the recirculation discharge valves, leaving only a very small bypass valve on each line open around the recirc pumps. The core water level decreased and a low-level alarm was experienced.

Eventually, the reactor operator re-established the feedwater flow and the recirculation flows. During this event, the core thermal margin was preserved, and it is significant that, while the reactor vessel water level was reduced for a period of time, the combination of the natural circulation of the system through the recirculation loop bypass valves and the isolation condenser operation made it possible for the reactor water level to recover.

The forgiving nature of the BWR design apparently made it possible for this event to be accommodated without serious consequence. Nevertheless, the event was a source of some concern because the operator took an erroneous action that affected core coolant inventory. The reactor was undamaged and, in fact, was ready to return to power the day after the event. The NRC, after some review, allowed the reactor to return to full power approximately one month later, on May 30th.

7.4.2 Lessons Learned

As with the Dresden 2 event (Section 7.1), there was a number of lessons learned at Oyster Creek, the most obvious of which is not to close all recirculation line discharge valves on a non-jet-pump plant.

7.4.3 Actions Taken

Although part of the existing procedures, it was necessary to clarify procedures and further instruct the operators so that this error would not occur again. The later BWR jet-pump plants, including GESSAR II, require no such procedures; their natural circulation flow path is through the jet pumps and entirely within the reactor vessel.

7.5 BROWNS FERRY 3 EVENT

7.5.1 Description

On June 1980, a manual scram of the Browns Ferry 3 reactor was attempted in conjunction with a planned shutdown for repair of a feedwater line in the turbine building. Aside from the need for this repair, plant conditions were normal.

The shutdown procedure involved first lowering the reactor power level to 36% by reducing the recirculation flow and inserting a number of control rods to decrease the neutron chain reaction; and secondly pushing the manual scram buttons to insert all control rods completely to terminate the neutron chain reaction. Complete control rod insertion is normally accomplished in less than 3 seconds after both scram buttons are pushed. In this incident, normal control rod insertion did not occur when the scram buttons were pushed.

Of 185 control rods, 10 were fully inserted prior to the manual scram. 77 rods failed to insert fully upon manual scram, with insertion ranging from position 02 (95% inserted) to position 46 (5% inserted). Observing this, the operator reset the scram; this procedure allows recharging of nitrogen-pressurized accumulators and draining of the scram discharge instrument volume. Manual scram was repeated. Insertion progressed somewhat, but 59 control rods remained only partially inserted. After a third reset the manual scram, 47 remained partially inserted.

Recharging and draining of the scram discharge instrument volume was repeated and the scram instrumentation automatically a fourth scram. All rods were now fully inserted, placing the reactor in normal shutdown condition. This was accomplished within about 14 minutes of the first scram.

7.5.2 Lessons Learned

Reference 1 concluded that appropriate steps should be taken by all BWR plants to guard against the following:

1. An obstruction in the SDV-SDIV connection pipes.

2. A configuration of the SDV-SDIV connector or vent pipe capable of producing a trip or loop seal. Such a trap or loop seal could possibly be the result of thermal expansion during hot conditions although it may not be present in a cold environment.

3. Interference by the CRW drain system with the operation of the SDV-SDIV system.

4. Failure of the SDV vent line valves to open.

5. Too slow drainage of the SDIV due to inadequate vent or drain capacity.

7.5.3 Actions Taken

A number of changes in the GESSAR II scram discharge volume have been made by adding redundant and diverse instrumentation to provide reliable scram. In addition, redundant vent and drain valves have been added to satisfy the single failure criterion.

7.5.4. References

1. Institute of Nuclear Power Operations and Nuclear Safety Analysis Center, "Analysis of Incomplete Control Rod Insertion at Browns Ferry 3," NSAC-20/INPO-3, December 1980.

8. EVOLUTION THROUGH TESTING

General Electric, has established a structured and disciplined approach to the development of product changes, both in identifying potential changes and in incorporating them into the design.

A design improvement passes through many steps on its way into a product offering. A design action list is maintained to assign priorities to identified design improvement opportunities. The front-end engineering is followed by development, prototype testing, qualification testing, manufacturing qualification, and final verification and design review - all overlaid with a heavy quality assurance program. In this manner assurance is obtained that only proven components and ideas are used, and that changes are well thought out and justified.

The following are a few of the tests conducted by GE to support evolution of the GESSAR II design. All told, more than 50 test facilities have been constructed and used to obtain design parameters and confirm design performance.

8.1 MATERIALS TESTING

The occurrence of intergranular stress corrosion cracking has affected the availability of operating plants. This cracking phenomenon is characterized as a non-ductile failure mode which requires three concurrent conditions: high stress, a susceptible material, and a sufficiently aggressive environment. If any of these three necessary conditions is absent, cracking will not occur. "Cracking" refers to microscopic intergranular cracks which typically cause leakage and not pipe rupture.

In stainless steel, cracking has been observed primarily in highly-stressed weld heat affected zones in piping, but it has also been observed in spargers and control rod drives. There was also a case involving Inconel Alloy 600 at a BWR/4 plant in 1973. The pipe cracking incidents represent a very small fraction - less than 0.6 percent - of the total number of nuclear system piping welds in the field.

To address this problem, GE embarked on a course which included design evaluation programs, comprehensive field surveillance, and laboratory research. A number of actions were taken to better define the conditions under which the cracking could occur. Extensive testing and examination led to the development of models and criteria for the selection and use of materials which are not susceptible. As a result of this program cracking mechanisms were identified and practical solutions which are backed by tests were found. A pipe test laboratory was built by San Jose to verify the theories and proof-test solutions.

Because of the rarity of the occurrence of cracks, it was necessary to have a facility which could provide a large statistical data base by testing a large number of specimens. Also, because the mean time to produce a crack in the field was relatively long (approximately 2 years), an accelerated testing method was needed. This facility, which began operation in 1977, is the world's largest pipe test laboratory. Over 1400 weld heat affected zones can be tested at one time. The tests can reproduce field cracking in about 100 hours, which makes possible the rapid statistical proof-testing of fixes. Improved materials have been tested for sufficiently long test periods to provide high assurance that they would well exceed the 40-year plant design life without cracking.

All materials used in BWR plants that are susceptible to intergranular stress corrosion cracking have been systematically evaluated against alternate materials. Through extensive tests, successful demonstrations have been made that welded Type 316 (nuclear grade) materials can operate in the BWR environment with large margins. This commercially available material, which controls carbon to less than 0.02 percent, is highly resistant to intergranular stress cor osion cracking.

For plants under construction which have Type 304 stainless steel, several methods were identified to reduce or eliminate susceptibility. A solution heat treatment process can be used for shop welds. The welded pipe is heated in a furnace and rapidly quenched. Another method is to apply a corrosion-resistant cladding under the weld zone before welding; this protects the weld zone from exposure to the oxygenated water. A third technique, called heat sink welding, uses water to cool the inside of the pipe during welding; this reduces the residual stresses which are a factor in the cracking mechanism.

An induction heat treatment stress improvement process has also been developed for use on in-place piping. This process produces compressive residual stresses on the inside diameter of as-welded piping.

8.2 SAFETY/RELIEF VALVES

There have been occasional instances of inadvertent reactor blowdown due to leaking pilot-operated safety/relief valves. Leakage in the pilot valve increases with time to the point where the main stage opens. Because of the magnitude of this leakage, the main stage stays open until the reactor pressure is reduced to approximately 60 percent of operating pressure. This problem has been addressed on operating plants by more frequent valve surveillance and maintenance, and by a new valve design.

On BWR/6, a direct-acting safety/relief valve is used, which does not contain a pilot operator. These valves are actuated in the relief mode by pressure transducers actuating a solenoid valve, and by a direct reactor pressure in the safety mode. Leakage cannot actuate these direct-acting valves.

Initial design and qualification tests on a safety/relief valve of the type to be used on BWR/6 included a 300-cycle open/shut life test, seismic tests, moment transfer tests, and actuator environmental tests.

8.3 MAIN STEAM ISOLATION VALVES

The rapid-closing capability of another important valve, and main steam isolation valve, was tested under simulated steam line break conditions at Commonwealth Edison's State Line steam plant. Long-term leakage characteristics are a separate phenomenon which is studied in the full-size MSIV test facility in San Jose.

8.4 FLOW CONTROL VALVES

The flow control valves in the recirculation system are new important features introduced first on BWR/5s.

From tests on the first recirculation flow control valves, problems with short bearing life and stem packing leakage were identified. The improved version was tested at the low flow valve test facility in San Jose, following tests done in the flow loops at Byron-Jackson Company in Los Angeles.

In this life test, each valve is housed in an enclosure which maintains plant environmental conditions external to the valve. The pressure, temperature, and chemistry of the water passing through the valve are maintained at plant conditions.

8.5 STEAM SEPARATORS AND CORE COMPONENTS

Test of full-size prototype BWR/6 separators under full steam flow conditions was completed in early 1974, and testing of actual production models was also completed in the test facility. This facility has also been used for testing feedwater spargers, in-core sensor tubes, and safety/relief valves.

8.6 FLOW INDUCED VIBRATION

Various problems caused by flow-induced vibration have occurred on light water reactor plants over the years. For the past several years, the NRC has required that "first of a kind" designs undergo preoperational and startup tests to ensure that vibration levels of key reactor internal components are within acceptable limits. A thorough visual inspection is performed on the reactor internals as part of the startup tests. General Electric performed startup vibration tests starting with Oyster Creek, Nine Mile Point 1 and Dresden 2.

For BWR/6, as part of the vibration qualification of reactor internals and to demonstrate sufficient design margins, full-scale tests were performed for components in the high flow hydraulic facility in San Jose. This facility, the largest of its kind in the world, was built in 1978 to conduct flow-induced vibration tests on vessel internals. It contains a full-scale 60° sector of a BWR/6 reactor vessel and can test actual production components. Vibration tests have been conducted on numerous components, such as jet pumps, ECCS coolant injection coupling components, control rod guide tubes and in-core sensors.

8.7 CONTROL ROD DRIVES

To ease the operator's job, BWR/6 has a solid-state rod control and information system which will move control rods in gangs of four, as well as individually. This is worth a 2-hour reduction in startup time. This ganged rod movement feature was tested at the control rod drive test facility in San Jose.

In conjunction with the ganged rod test, a design qualification test of the control rod drive pump which supplies the high pressure water that powers the drives was also performed. This pump is identical to those used for BWR/6.

Testing of control rod drives has been an ongoing activity since 1957. In recent years, the test facilities in San Jose have tripled and the test hours have quadrupled. The drives have a high demonstrated scram reliability. In operating plants, there have been, at most, three malfunctions in over 230,000 individual drive scrams.

The BWR/6 scrams faster to improve thermal margins during operational transients. To accommodate the increased speed of insertion, the drive was strengthened to improve its structural stability. Full-scale testing was performed under normal and transient operating conditions, and incorporated cyclic tests up to six times design lifetime.

The hydraulic drives require periodic seal replacement and checks. Special servicing equipment, which is located beneath the reactor pressure vessel, has been made semi-automatic. Also, the new equipment is driven by air motors rather than the electrical drives used in current operating plants, which should help in this high humidity undervessel area.

In current operating BWRs, removal and replacement of a drive typically takes about 3 hours and requires a crew of three or four workers plus a lead technician. With the BWR/6 equipment, the job can be done in one hour by a crew of one worker and one lead technician. Maintenance exposure is reduced on two counts - less time and fewer people.

Even shorter times have been demonstrated at the GE undervessel mockup test facility in San Jose, including tests under plant conditions such as having the workers outfitted in waterproof suits and double gloves. At this facility, control rod drive changeout equipment for the first few BWR/6 units was tested for operability before being sent to the field for installation. Incidently, several facilities like this one are also used to train GE and utility plant maintenance crews.

8.8 IN-CORE NEUTRON SENSORS

The in-core sensors, which are located in tubes throughout the core, see very high neutron fluxes in operation. Several typically need replacement every refueling. For plants prior to BWR/6, replacement of in-core neutron sensors is performed from above the core, and is therefore on the refueling critical path. On BWR/6, the sensors are replaced from below the reactor, this activity from the critical path. The irradiated detector is removed using new remotely operated semi-automatic equipment; the fresh unirradiated detector is inserted by hand.

In addition, a sensor with a projected three-fold longer life has been developed. For pre-BWR/6 plants, where replacement is on the critical path, conversion to the new sensors results in reduced critical path servicing time. Although replacement will not be on the critical path for BWR/6 plants, BWR/6 will benefit from the longer-life sensor through reduced maintenance.

The undervessel mockup test facility is being used for shakedown testing of the semi-automatic system that removes the detectors and chops them up into pieces for disposal into a shielded cask.

8.9 INCLINED FUEL TRANSFER

For the Mark III containment, fuel transfer between the Fuel Building and the Reactor Building is through an inclined transfer tube. A full-scale test stand was constructed in GE facilities to test the prototype equipment.

This program includes a prototype test, a 40-year life test, and an installed site evaluation. The completed prototype test confirmed system operability. The life test has passed the equivalent of more than 30 years of performance.

8.10 SEISMIC QUALIFICATION

To confirm the ability of safety-related equipment to remain function through an earthquake, a seismic simulator was constructed and used for equipment testing. The seismic testing table can operate along two axes simultaneously, at an acceleration in excess of 3 g. The 9 by 13-foot shake table permits testing of full-sized control panels.

Shake tests were conducted on an 8x8 fuel assembly and a variety of equipment, including control rod drive hydraulic units, ECCS pump motors and main steam isolation valve actuators. These tests confirm the accuracy of seismic calculations and qualify the equipment.

8.11 ENVIRONMENTAL QUALIFICATION

The Environmental Test Facility was set up specifically to conduct the environmental portion of testing required for the qualification of Class 1E items. Environmental qualification of all BWR safety related items is conducted in these facilities. Testing has included small piece parts such as pressure switches, connectors, cable, PC boards, to large, more complicated items such as Optical isolators, valve control monitor panel and the NSPS panel.

8.12 PRESSURE SUPPRESSION CONTAINMENT TESTING

To confirm the analytical models used to define hydrodynamic loads produced in the Mark III suppression pool, GE constructed a pressure suppression test facility which duplicated in volume an 8° segment of the Mark III containment and suppression pool.

Steam is taken from a flash boiler through a 52-foot-high suppression chamber. A scaled drywell, 11 feet in diameter by 30 feet high, is instrumented to measure the effectiveness of the pressure suppression.

Full-scale and 1/3 area scale single-vent, and 1/9 area scale multi-vent test sections have been employed to confirm design loads arising from such phenomena as pool swell, air clearing and chugging. All planned Mark III confirmatory testing has been completed.

This facility has been adapted to Mark II testing by using the same blowdown vessel, blowdown line, and drywell with a separate Mark II suppression chamber mockup. The Mark II portion was added to the facility in 1975 and was later modified to perform tests to confirm Mark II condensation oscillation loads.

8.13 ECCS TESTING

General Electric also has a wide variety of test facilities devoted to the ECCS. The objective is to determine actual performance characteristics. These tests have included core spray distribution, counter flow tests, core cooling and heat transfer tests, and a wide variety of tests to support the 10CFR50 App. K calculations.

Full-scale, electrically heated, simulated reactor fuel bundles are used in ATLAS. The test loop itself creates a closely simulated reactor environment. Operating conditions, to include pressure, flow and temperature are established under both steady-state and possible transient conditions. To detect boiling transition (rod surface overheating), thermocouples monitor the simulated fuel rods at many locations. They help determine the point at which highly efficient boiling heat transfer begins to deteriorate. When the onset of transition boiling is detected by one of the thermocouples, the data acquisition system is activated. Single- and two-phase pressure drop characteristics of rod bundles, as well as other fuel-related components, also can be measured.

8.14 SUPPRESSION POOL SCRUBBING TESTS

Suppression pool scrubbing tests were conducted in 1982 at GE's suppression pool hydrodynamic test facility at Vallecitos to quantify the fission product retention capability of the BWR suppression pool during postulated severe reactor accidents.

The suppression pool scrubbing tests demonstrated the capability of the pool as an effective fission product trap. The program developed at first principle analytical model and generated sufficient test data to verify the model.

The key breakthrough of the test program was the demonstration of bubble hydrodynamics. A bubble "shattering" phenomena was observed and shown to be consistent with theory. The shattering of large bubbles into small stable bubbles allowed direct application of the model for all BWR geometries independent of scale. The model predicts that the proof would retain essentially all non-gaseous fission products which might be released in a postulated core meltdown accident. Therefore, even for accidents much more severe than the Three Mile Island accident, the health and safety of the public would be maintained by the fission product trapping in the suppression pool.