The Effects of Regulation on the Performance of Nuclear Power in the United States and The Federal Republic of Germany

by

Seth David Hulkower

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SETH DAVID HULKOWER

B.S. Mech. Eng., Tufts University
(1981)
B.A., Tufts University
(1981)

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Signature of Author ______________________
Technology and Policy Program
May 9, 1986

Certified by ______________________
Kent F. Hansen, Dept. of Nuclear Engineering
Thesis Supervisor

Accepted by ______________________
Richard DeNeufville
Chairman, Technolgy and Policy Program

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ABSTRACT

The nuclear power industry of the Federal Republic of Germany has achieved consistently better reactor operating performance than has the U.S. industry. Earlier work has suggested that a major source of the difference is in capacity factor losses caused by regulatory practices. An investigation of the problems attributed to regulation in the United States, which caused losses from 1975 to 1984, was performed. Fifteen major issues were identified, which comprised 85 percent of all regulatory losses. The performance of the German industry then was analyzed to discover differences in regulatory practices.

Most of the U.S. regulatory losses were found to be associated with steam generators, reactor coolant systems, and containment systems. The regulatory losses in the Federal Republic of Germany included the retraining of a plant staff after an accident, a long-term derating of a plant because of inadequate backup safety systems, and several smaller problems. The German industry applies inspection and repair standards for steam generators that equal or exceed U.S regulations, and it treats these losses as part of normal plant maintenance. However, the German industry also assigned the largest single cause of capacity loss in BWRs--recirculation pipe replacement--to the voluntary maintenance category because they weren't strictly ordered to shut down, only pressured with the threat of stringent inspection standards. When the BWR pipe replacement outages were added to the Federal Republic of Germany's total regulatory loss, and the steam generator losses were subtracted from the U.S. total, the Federal Republic of Germany was found to have greater regulatory losses. It can therefore be concluded that the sources of poor U.S. performance relative to the Federal Republic of Germany come from areas other than regulation.

Thesis Supervisor: Dr. Kent F. Hansen
Title: Professor of Nuclear Engineering

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S. D. H.

May 1986

West Berlin

To Charles Edwards,
my grandfather.

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Chapter 1

1.0 Nuclear Power Performance

1.1 Introduction

The U.S. commercial nuclear power program has been operating for over twenty-five years. At the end of 1984, there were 77 large light water reactor (LWR) nuclear plants in commercial operation, each having a generating capacity of 400 megawats electric (MWe) or more. About three-fifths of these plants had been in operation for at least ten years. Despite this wealth of experience, the U.S. industry has failed to match the performance records of the nuclear industries of several other countries.

The purpose of this project is to examine some of the reasons for the differences in performance between LWRs in the United States and the Federal Republic of Germany. It is a follow-on to a similar study of the industries in the two countries which identified losses attributed to regulation as one of the chief differences in performance. [1]

In the initial work, the measure of performance used was the capacity factor, the energy produced by a given plant in a year divided by the energy which would have been produced if the plant had been running at full power for every minute of the year. Plant capacity factors in the United States and the Federal Republic of Germany from 1980 to 1983 were compared. The results showed the U.S. pressurized water reactors (PWR) performing at an average capacity factor of 58.3 percent and the FRG PWRs at 77.1 percent. The capacity losses were characterized as scheduled outages, forced outages, and regulatory outages. Over 40 percent of the overall performance difference between the two countries arose from the difference in the regulatory category.

While the U.S. industry attributed 7.7 percent of total capacity losses to regulation, the German industry credited regulation with less than 0.1 percent of its losses. This report focuses in particular on the origins of these regulatory differences.

The Federal Republic of Germany is only one of several countries which employ light water reactor technologies similar to the United States. The FRG program is, however, older than most others and has a large enough number of both boiling and pressurized water reactors to make relevant comparisons.

1.2 The Cost of Poor Performance

Nuclear power plants in the United States have a lower operating cost than all other base load supply systems except hydro. Hydro power is not available in many parts of either the United States or the FRG and thus those utilities which own nuclear plants would generally like to operate them as much as possible. When a nuclear plant is shut down, a utility must either run its more expensive power plants or purchase power from another utility to meet its demand. The cost of replacement power varies across the country and with the seasons. It is also strongly affected by oil prices. A rough figure for the cost of replacement power is $1,000/MWe-day. Thus, each day that a 1,000 MWe plant is out of service costs $1,000,000 in purchased power. At the end of 1984 there were 65,000 MWe of nuclear power capacity installed. A one-percent improvement in the performance of the nuclear industry would reduce the need for purchased power by 234,000 MWe-days per year, or $234,000,000 per year.

The value of even a small improvement in the performance of the U.S. industry is thus apparent and is the justification for this research. Any differences in operation which could push the U.S. capacity factors towards those of the FRG without reducing safety are worth hundreds of millions of dollars each year.

1.3 Scope of the Study

This study examines the performance of all light water reactors with net ratings above 300 MWe from the beginning of 1975 until the end of 1984 in the United States and the FRG. This period was chosen because of the availability of performance data, because it spanned the accident at Three Mile Island and might show any effects which regulation had upon performance from that event, and because it provided a large enough set of data points from both countries to make relevant comparisons. A study reaching further back in time would find very few German plants to compare with the United States. Two German plants which came into commercial operation in late 1984 have not been included.

1.4 Outline of the Thesis

This report begins with a description of the structures of the American and German nuclear power industries in Chapter 2. The regulatory process in the two countries is reviewed as part of this general description.

The sources of U.S. capacity loss which have been attributed to regulation are examined in Chapter 3. The U.S. utilities have reported all their outages in great detail and in this chapter all the regulatory outages for each plant are sorted to determine the problems which have led to regulatory loss.

The U.S. industry's perceptions of the regulatory issues are examined in Chapter 4, which includes the results of a series of interviews conducted with U.S. industry officials. The latter provide additional insight into the history of these problems as well as an understanding of the interactions between members of the industry.

A parallel analysis of FRG industry experience is presented in Chapter 5. A comparison of the problems encountered in the German and U.S. industries is made, focusing on problems which arose in both countries, including causes of regulatory losses in the United States, which were not attributed to regulation in the FRG.

Finally, differences in regulation and operating practices and their effect on performance are discussed in Chapter 6. Differences in the nature of responses to problems are highlighted including issues which were problems for only one country. In addition, several proposals for changes in U.S. industry practices are made, along with recommendations for future work.

Chapter 2

2.0 Industry Structure

2.1 United States

2.1.1 Nuclear Power Generation

In 1975, twenty-three utilities operated thirty-eight nuclear plants with a capacity of 27,865 MWe that accounted for 9 percent of total electricity generation. [2][3] By 1984, thirty-seven utilities operated seventy-seven plants with a capacity of 65,049 MWe that provided 13.6 percent of total electricity generation. The breakdown for each of the years of the study period is listed in Table 2.1. A list of all the U.S. plants in the study is given in Appendix 1.

2.1.2 Safety Regulation

In the United States, regulation of nuclear power plant safety is a Federal responsibility. The Nuclear Regulatory Commission (NRC) has been empowered by the United States Congress under the Energy Reorganization Act of 1974 and the Atomic Energy Act of 1954 with ensuring the safe operation of civilian nuclear power. Towards this end, the NRC is "authorized to conduct such reasonable inspections and other enforcement activities as needed to insure compliance. . " with safety regulations. [4]

2.1.3 Economic Regulation

The economic regulation of the investor-owned utilities is performed within each state by an agency which henceforth will be referred to as a Public Utility Commission (PUC). (Although the name may vary from state to state, the purpose is roughly

LWR's IN THE UNITED STATES

YEAR	No. of Plants	No. of Utilities	Total Megawatts	Percent of Total Generation
1975	45	28	33,938	9.0%
1976	49	30	37,689	9.4
1977	57	34	44,928	11.8
1978	60	35	47,841	12.5
1979	62	35	49,531	11.4
1980	63	35	50,443	11.0
1981	68	35	55,622	11.9
1982	69	35	56,770	12.6
1983	72	36	59,726	12.7
1984	77	37	65,049	13.5

Sources: Columns 2, 3, and 4 from the INPO database.
Column 5, Energy Facts 1984, Energy Information Administration, US Department of Energy, DOE/EIA-0469, May 1985, p. 38.

Table 2.1

identical.) The utilities have been granted local monopolies to provide electricity, and the PUCs have the right to regulate electricity prices to prevent monopolistic pricing. In return for the price regulation, each utility is assured a fair rate of return on all prudent capital investments and allowed to recover all reasonably incurred costs.

During the 1950s and 1960s the cost of electricity steadily declined as utilities kept building larger power plants and achieving greater economies of scale. The PUCs and the utilities worked together to set rates and drew little attention because the costs were falling. All this changed in the early 1970s when oil shortages led to sharply higher prices. Then, utilities began to apply for rate increases nearly every month, and the rate hearings became the focus for consumer groups angry over these price rises. The PUCs came under intense pressure to hold prices down and this led to a change from a cooperative to an adversarial relationship between the PUCs and the utilities. [5] This pressure to hold down prices has eased little since the early 1970s and the adversarial relationship still exists across most of the United States.

2.1.4 Industry Collaboration

The U.S. nuclear power industry relies upn several industry organizations for technical support, including, most notably, the Institute of Nuclear Power Operations (INPO) and the Electric Power Research Institute (EPRI). EPRI is a research organization which receives its funding on a voluntary basis from the U.S. utilities and performs studies in areas of interest to those utilities, including nuclear power plant technology. INPO is dedicated solely

US LIGHT WATER REACTOR
NUCLEAR STEAM SUPPLY VENDORS
(as of 31 December 1984)

Vendor	Type	Total
Babcock and Wilcox	PWR	9
Combustion Engineering	PWR	11
Westinghouse	PWR	32
General Electric	BWR	25
Total PWR		52
Total BWR		25
All LWR		77

Table 2.2

to nuclear power. Most of its employees are from nuclear utilities, and stay with INPO for two years, afterwards returning to their jobs in the industry. INPO's task is to support operations by collecting information on the operation of the nuclear plants and sharing this information with all utilities so as to help them achieve high levels of performance.

2.1.5 Manufacturers

In the United States, there are four manufacturers of large light water reactor nuclear steam supply systems (NSSS): Babcock and Wilcox, Combustion Enginering, and Westinghouse Electric, which produce pressurized water reactors (PWRs); and General Electric (GE), the only boiling water reactor (BWR) manufacturer. The industry is dominated by Westinghouse and GE, as can be seen by the data listed in Table 2.2. Because there have been no new orders recently, these manufacturers now concern themselves primarily with providing services for plants in operation.

2.1.6 Design and Construction

With few exceptions, the U.S. utilities have gone outside their own staffs for the design and construction of their nuclear plants, relying on firms known as Architect/Engineers (AEs) to do the detailed engineering work for all the non-NSSS equipment. The contracting utility may then hire another AE to perform the construction or may manage construction itself. The plants in the study have used nine AEs and eleven constructors, while six of the utilities have provided at least a part of the architectural engineering services and eleven have served as the construction manager.

2.2 Federal Republic of Germany

2.2.1 Nuclear Power Supply

At the beginning of the study period in 1975, three large nuclear plants were operating with a total capacity of 1,610 MWe. Several smaller LWRs and non-LWR plants were also in service, and together these nuclear plants accounted for 9.2 percent of total electricity generation. [6][7] By the end of the study period in 1984, eleven large LWRs were in service with a capacity of 9,798 MWe and the nuclear industry provided 27.6 percent of total electricity generation. [8] This breakdown for each of the years of the study is listed in Table 2.3.

The utilites are all investor-owned, but the predominant investors are the land (or state) governments. Appendix 2 lists the German plants in the study.

2.2.2 Safety Regulation

Until 1955, the FRG was prohibited by the Western Allies from developing nuclear power. By the end of 1959, the FRG's Atomic Energy Act was enacted, [9] and the German Constitution was amended to allow Federal laws to stipulate that, with the approval of the Federal Council, the states will enforce designated laws. The Atomic Energy Act of the FRG took advantage of this change to set out the guidelines for nuclear power and then charge the states with enforcement. Each state holds the responsibility for the safety of the operating plants.

NUCLEAR POWER IN THE FEDERAL REPUBLIC

Year	Nuclear Generation (GWh)	Total Generation (GWh)	Percent Nuclear Generation
1975	21,864	238,456	9.2%
1976	24,348	267,613	9.1
1977	35,153	268,760	13.1
1978	35,008	238,569	12.3
1979	41,609	298,644	13.9
1980	42,619	298,494	14.3
1981	52,492	301,574	17.4
1982	62,448	303,256	20.6
1983	64,660	310,935	20.8
1984	91,444	331,187	27.6

Sources: "Die Öffentliche Elektrizitätsversorgung" 1982 and 1983, Vereingung Deutscher Elektizitätwerke, Frankfurt.

"Elektizitätswirtschaft", 1985, Vereingung Deutscher Elektizitätwerke, Frankfurt.

Table 2.3

Within the Federal government, the Federal Minister of the Interior (Bundesminister des Innern or BMI) is responsible for the regulations promulgated under the Atomic Energy Act. The Federal government and the state governments then rely upon several private organizations to draft regulations and oversee their implementation.

2.2.2.1 The Nuclear Safety Standards Commission - KTA

In 1972, the Federal Minister of the Interior established the Nuclear Safety Standards Commission (Kerntechnischer Ausschuss or KTA) to bring together all the participants in the nuclear industry with sufficient expertise in nuclear power to develop safety standards. Five groups of ten members each are represented on the KTA: the manufacturers and constructors, the owners and operators, independent experts, Federal authorities and state authorities, and organizations with special technical knowledge.

The KTA meets in task groups which draft the safety regulations. The drafts are reviewed by KTA subcommittees and then issued for three months for public comment. After the regulation has been finalized it must then be approved by a 5/6 majority of the KTA. Thus, if one of the five member groups is opposed to it, the regulation will not pass. Although an approved regulation is not law, failure to comply with it imperils the plant's license. [10]

2.2.2.2 The Reactor Safety Commission - RSK

While the BMI waits for the KTA to agree upon regulations, it relies upon the Reactor Safety Commission (Reaktor Sicherheitskommission or RSK) to provide guidelines on the design, construction, and operation of nuclear power plants. The RSK has twenty members, all of whom are personally appointed by the Federal Minister of the Interior. The members are chosen from the following fields: reactor operations; civil and mechanical engineering; thermodynamics; chemical engineering; materials; construction; instrumentation and controls; reactor physics; electrical engineering; reactor chemistry; radiation protection; environmental protection; radiation biology; and nuclear medicine.

The appointments are personal and voluntary. Each member represents not the organization for which he or she works but the expertise for which he or she is chosen. The RSK guidelines do not enjoy the full weight of law but are used for reference by the BMI and the states while the KTA develops its regulations. [10]

2.2.2.3 The Reactor Safety Association - GRS

The Reactor Safety Association (Gesellschaft fur Reaktorsicher-heit or GRS) is one of the independent experts used by the BMI and the states. It peforms technical studies on the safety of nuclear facilities and radiation protection, and participates in the formulation of guidelines and regulations by the RSK and KTA. Upon request by the government agencies, the GRS undertakes analyses of specific safety issues. The GRS is responsible for the management of the German light water reactor safety research program. [10].

2.2.2.4 The Technical Inspection Agencies - TÜVs

There are eleven Technical Inspection Agencies (Technische Uberwachungs-Vereine or TÜV) in Germany and each one of them is a private, independent company. The TÜVs have existed for over one hundred years, serving as independent inspectors for industry. They are similar in nature to Underwriters Laboratories in the United States but much broader in scope, performing inspections of equipment ranging from pressure vessels to motor vehicles. Seven of the eleven TÜVs have departments devoted to nuclear power. The TÜVs perform inspections and tests of plants during construction and operation. [10]

2.2.3 Economic Regulation

Regulation of the price of electricity in the FRG is handled in each state by the Ministry for Trade and Commerce. The ministry is expected to review the costs of applying electricity and then establish a rate structure to cover the costs and provide a fair rate of return. The utilities are able to negotiate private contractors with large industrial customers and these contracts are not subject to review by the ministry. The cost reviews performed by the ministries consist only of a verification of the actual costs and that the costs were consistent with other projects of similar scope.

2.2.4 Industry Collaboration

In addition to the research provided to the industry by GRS, the German electric utility industry relies upon the work of the

Association of Large Power Plant Operators (Vereinigung der Grosskraftwerksbetreiber or VGB) for research on performance. The VGB is a predominantly German organization, which, however also includes members from most of the European nations as well as the United States, Brazil, Argentina, South Africa, India, and Australia. It has a small permanent staff and many large committees which meet only a few times a year. The committees are charged with the task of studying specific problems and preparing recommendations on the solution of these problems. Like EPRI, their work is not limited to nuclear power, but covers all areas of power production. [11]

2.2.5 The Manufacturers

Over the years there have been four vendors of light water NSSS in Germany, but now there are only two. In the early years, Siemens and AEG competed for orders, with Siemens offering PWRs and AEG BWRs. In 1969, these two companies began to merge their nuclear operations into a new company named Kraftwerk Union (KWU), and the last stages of the merger were completed in 1973. The pace of the merger was dictated by licensing agreements which the parent firms held with Westinghouse and General Electric. Another company is Brown-Boveri which has designed only one plant that has not yet come into service. Thus, all the plants in this study have been manufactured by either KWU or its parents.

KWU owns very little of the actual manufacturing equipment and has little capital invested in the nuclear industry. The manufacture of almost all parts of the NSSS are subcontracted out.

2.2.6 Design and Construction

The design and construction of plants in the FRG has been handled almost exclusively by KWU. In the instances where KWU wasn't the sole NSSS and, turbine-generator manufacturer, AE, and Constructor, the job has been done by its parents AEG or Siemens, or begun by them and completed by KWU.

All plants were turn-key in the sense that the utilities did not accept ownership of a plant until it had been operating uninterrupted at full power for a month. The costs were fixed with the agreement that the contract would be renegotiated in the event of changes in regulation. The contracts also stipulated that KWU would pay a percentage of all changes whether requested by the utilities, demanded by the regulators, or recommended by its own staff. The KWU contracts have included performance guarantees for the first two years of operation. Barring operator errors, KWU would pay penalties for a plant with low availability. The agreement for the Krummel plant guarantees 70 percent energy availability in the first year and 75 percent in the second year.[1]

[1]Krummel began commercial operation in late 1984 and was not part of this study.

Chapter 3

3.0 U.S. Data Analysis

The S.M. Stoller Corporation is conducting an ongoing study of nuclear power performance for the Electric Power Research Institute (EPRI). Using the Monthly Operating Reports submitted to the NRC, supplemented by the NRC Gray Books, Licensee Event Reports (LERs) 'Nuclear Power Experience', technical papers, and contacts with the operators of individual units, they have compiled the Operating Plant Evaluation Code (OPEC-2) data base. [12] OPEC-2 describes every outage or derating at a U.S. plant over 400 MWe, providing explicit information on the system and component responsible for the outage. It also lists any external events which led to the outage, along with a brief written description of the outage. The external events include operator error, preventive maintenance, and several regulatory categories. Appendix 3 shows the level of coding and categories within each level. More recently, INPO has assumed responsibility from EPRI for maintaining the OPEC-2 database.

The OPEC-2 file was made available to MIT by INPO for this project subject to the condition that no plant or utility be specifically identified. The data has been used to identify areas of significant capacity loss across the industry.

3.1 Data Analysis

3.1.1 Selection of Regulatory Outages

The data was sorted for all events with a regulatory coding as the external cause of event. There were two additional groups of events which had not been coded as regulatory in OPEC-2 but were judged to be regulatory in nature. These were certain fuel limitations relevant to plant safety and intake and discharge water restrictions. The fuel limitations were considered regulatory because they resulted from regulations which restricted the operation of the plant. In addition, the intake/discharge water restrictions are based upon regulations of the Environmental Protection Agency (EPA). They limit flow rates and temperature differentials to prevent damage to fish and plant life near the power plant. All the codes used to identify regulatory events are shown in Table 3.1.

Using these two sets of criteria for sorting during the study period, 5,102 events were identified as regulatory out of a total of 37,492 events in the OPEC-2 file.

3.1.2 Preliminary Analysis

The first analysis of the regulatory data was by major system so as to determine which systems most often had regulatory problems. The hours lost by each system for all U.S. plants in each year were calculated and then divided by the total hours for these plants in each year. The result was an average capacity loss factor for each system for each year.

The results of this analysis are listed in Table 3.2, they show two things: that a pronounced increase in the regulatory capacity losses began in 1979; and that most of the problems occurred in the Containment, Steam Generator, and Reactor Coolant systems, and the Undefined category. The contribution of the two TMI units has been subtracted from the yearly totals to show that while their contribution to capacity loss has been significant, it has not been

REGULATORY OUTAGE CODES
OF THE OPEC-2 DATABASE

All Events with the External Influence Descriptions:

NRC Originated
- o Regulatory/Operational limit (Safety Limit of Tech Spec)
- o Regulatory requirement to inspect for possible deficiency
- o Regulatory requirement to modify equipment due to malfunction or construction/design deficiencies
- o Regulatory requirement to modify equipment due to more restrictive criteria
- o NRC licensing proceedings and hearings
- o Unavailability of safety-related equipment

Additional categories added to the sort:

Fuel and Core, Safety Restrictions
- o ECCS peaking factor (PWR)
- o EOL scram reactivity/rod worth restrictions (includes shutdown margin
- o Core tilt/Xenon restriction (out of flux band)
- o BWR thermal limits (includes "rod limited")
- o Thermal power restrictions
- o Reactivity coefficient (e.g., mod. temp. coeff.)

Circulating Water/Service Water System, Intakes/discharges
- o Excessive fish kill
- o EPA discharge limit

Table 3.1

REGULATORY CAPACITY LOSSES FOR ALL SYSTEMS
1975-1984
(in percent)

	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Undefined*	-	-	0.001	0.006	2.208	4.199	3.559	2.917	2.883	2.753
Fuel and Core	3.082	1.555	0.748	0.562	0.227	0.132	0.075	0.148	0.143	0.164
Reactor Coolant	0.782	0.152	0.259	1.628	1.871	0.390	0.130	0.798	4.022	6.861
Steam Generator	0.635	1.843	0.761	0.857	0.603	1.426	2.632	3.753	1.576	2.362
Feedwater	-	-	0.023	0.012	2.141	0.173	0.026	0.027	0.260	0.274
Turbine	0.035	0.101	0.066	0.053	0.304	0.224	0.054	0.049	0.038	0.044
Electrical System	0.004	0.070	0.156	0.189	0.037	0.065	0.003	0.037	0.046	0.069
Auxiliary System	-	0.006	-	0.002	0.002	0.177	0.016	0.003	0.003	0.004
Refuel. and Maint.	0.021	-	0.020	0.226	0.176	0.256	0.118	0.217	0.223	0.253
Circ Water	0.028	0.056	0.118	0.192	0.101	0.063	0.029	0.018	0.043	0.050
Core Cooling	0.050	0.105	0.002	0.163	1.360	0.400	0.177	0.131	0.988	0.434
Containment	0.288	0.548	0.687	0.479	4.378	6.957	3.005	4.857	3.579	2.749
Structures	-	0.563	-	1.020	0.151	0.105	-	0.072	-	0.086
Chem Vol Control	-	0.078	-	-	0.028	0.357	-	0.006	0.120	0.097
Reactor Trip Sys	0.126	0.030	-	-	-	0.021	-	0.001	0.272	0.012
Condenser	0.048	-	-	0.011	0.003	0.010	0.002	-	0.053	-
Other	0.007			0.129	0.008	0.003	0.007		0.007	0.095
Total	5.106	5.107	2.841	5.529	13.598	14.958	9.833	13.034	14.256	16.307
TMI 1 & 2					1.243	3.187	3.069	2.916	2.850	2.640
Total w/o TMI	5.106	5.107	2.841	5.529	12.355	11.771	6.764	10.118	11.406	13.667

* Nearly all Undefined losses came from the two TMI units.

Table 3.2

the only factor in the increased capacity losses. "Undefined" are events for which the utilities did not specify a system, and this was the category into which most of the TMI losses fell. The "Other" category includes all plant systems which individually had a very small influence on regulatory capacity loss.

The data was further separated into Containment, Steam Generator, Reactor Coolant, and Undefined files and each reviewed separately.

3.2 System by System Analysis

The task when analyzing the specific systems was to identify the problems in each system which caused the losses. The losses attributed to Containment, Steam Generator, Reactor Coolant, and TMI 1 & 2 during the ten year study period are shown in Table 3.3 as percentages of total regulatory loss. They account for nearly four-fifths of all regulatory losses.

The losses in each system are further broken down by the external events and are shown in Tables 3.4 (a), (b), and (c). The Containment was most often influenced by inspections and modifications for deficiencies. The "all of the above" category was specified most frequently for the Reactor Coolant system losses. The losses in the Steam Generator were blamed on violations of the technical specifications and inspections.

3.2.1 Containment Outages

Over half of the Containment outage hours were attributed to modifications for deficiencies and a review of the hours lost was

PRINCIPAL SYSTEMS CONTRIBUTING TO
REGULATORY CAPACITY LOSS
(percent of total regulatory losses)

System	Percent
Containment	27.7%
Reactor Coolant	17.8
Steam Generator	16.2
TMI 1 & 2	16.6
	78.3%

Table 3.3

carried out to determine what these deficiencies were and what the corrective action was. As can be seen in Table 3.2 the problems began in 1979, peaked in 1980 and have declined somewhat since then. The outage hours were attributed to modifications to the Torus in BWRs and a group of Inspection and Enforcement Bulletins (IEBs) related to seismic analysis of safety system piping. The next two subsections will describe these issues in some detail. The precise losses resulting from these causes are shown in Tables 3.5 and 3.6.

3.2.1.1 Torus Modifications

In 1975 General Electric recognized that some of the hydrodynamic loads of the Mark I containment had not properly been taken into account. The NRC, with industry compliance, established a two-track approach to solving the problem. A short term program was begun to determine the extent of the inadequacies of the earlier design and to identify any serious safety problems. The long term program was to develop recommendations for the permanent resolution of the deficiencies. The long-term study was due in 1979 and was issued as NUREG-0660, a description of the acceptable resolutions. The implementation of these solutions began in 1979. The short term program had virtually no effect on operations. All the losses were caused by the modifications to the torus recommended by the long term program.

3.2.1.2 Seismic Analysis Bulletins

Three seismic analysis bulletins were issued:

(a) IEB 79-02: Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts.

This bulletin applied to all operating licensees and holders of construction permits and was issued after an operating plant, Millstone Unit 1 in Waterford, CT, discovered the structural failure of some of its pipe supports and, further, that some of those still intact had not been properly tightened. Deficiency reports filed at Shoreham in Brookhaven, NY, a plant under construction, indicated that "design of base plates using rigid plate assumptions has resulted in underestimation of loads on some anchor bolts. Initial investigation indicated that nearly fifty percent of the base plates could not be assumed to behave as rigid plates." The bulletin directed all licensees and permit holders to verify that the flexibility of the base plates was taken into account in the calculation of the anchor bolt loads, that a sufficient margin of safety existed between the bolt design load and bolt ultimate capacity, and that the design requirements for cyclic loading had been met. [13]

(b) IEB 79-14: Seismic Analyses for As-Built Safety-Related Piping Systems

This bulletin was issued to all licensees for action and permit holders for information. It indicated that the analysis of piping systems had been performed at some plants with drawings that did not match the installed configuration. Licensees were ordered to inspect the piping, prepare precise as-built drawings (drawings which showed the precise installed piping configuration), recalculate all seismic loads, evaluate the non-conformances, and correct as needed. [14]

CHARACTERIZATION OF EVENTS BY SYSTEM

Total Plant Hours 1975-1984: 5,319,601.0
Total Regulatory Loss Hours: 568,226.3 = 10.7%
Total Regulatory Events : 5,102

Containment (157,584.9 Hours, 462 Events)

Event	Hours	No. of Items	%Hours	Hours/ Item
Tech Spec Viol.	1971.6	44	1.3	44.8
Inspections	34228.8	192	21.7	178.3
Mod. for Deficiency	88473.0	166	56.1	533.0
Mod. for Regulation	20215.8	36	12.8	561.6
Licensing	12741.9	18	8.1	707.9
		458		

(a)

Reactor Coolant System (101225.6 Hours, 633 Events)

Event	Hours	No. of Items	%Hours	Hours/ Item
Tech Spec Viol.	7670.0	156	7.6	49.2
Inspections	20573.4	308	20.3	92.9
Mod. for Deficiency	16094.6	45	15.9	357.6
Mod. for Regulation	10189.2	21	10.1	118.6
All of the Above	46391.8	99	45.8	468.6
		629		

(b)

Steam Generator (92208.9 Hours, 264 Items)

Event	Hours	No. of Items	%Hours	Hours/ Item
Tech Spec Viol.	40598.5	133	44.0	305.3
Inspections	44510.2	119	48.3	374.0
Mod. for Deficiency	6980.2	11	7.6	634.6
		263		

(c)

Table 3.4

(c) IEB 79-07: Seismic Stress Analysis of Safety-Related Piping

By this bulletin, the NRC directed five power plants to shut down when a review of seismic analysis computer codes determined that a code in use did not properly evaluate the loads during a seismic event. The bulletin ordered all licensees and permit holders to identify the calculation methods used in the computer analyses, provide complete listings of the pertinent sections of the computer programs, and verify that the results were checked against "benchmark problems or compared to other piping computer programs." [15]

3.2.2 Reactor Coolant System

Nearly half of the hours lost in the Reactor Coolant System (RCS) fell into the category "all of the above", as can be seen in Table 3.4(b). The utilities did not identify only one type of NRC event as the cause. Another quarter of the outages were described as inspections. When investigated, the inspections covered virtually all the components in the RCS and ranged from isolation valve tests which lasted for less than an hour to ten year in-service inspections of the reactor vessel lasting several months. The "all of the above" category events, however, had to do with a single issue, Intergrannular Stress Corrosion Cracking (IGSCC).

This cracking occurred in an unstabilized austenitic stainless steel designated Type 304. The steel had been recognized as far back as the 1920s as being susceptible to IGSCC, but only under highly corrosive environments and conditions of high stress. The water in the recirculation system of a BWR is of high purity; it is demineralized, cleaned of organic material, and its pH is controlled within a narrow range. The stresses and corrosive environment known to cause IGSCC were believed to be absent in BWRs when the 304 steel was chosen.

3.2.3 Steam Generator Losses

The Steam Generator losses were evenly split between violations of the technical specifications and inspections as shown in Table 3.4(c), but a close review of the descriptions of the hours lost indicated that many of the hours in each category were actually for tube repairs and replacements. These were problems discovered during a scheduled inspection or when a violation of the technical specificaton necessitated a shutdown. These outages lasted anywhere from several months up to a year, during which time some of the steam generators were completely replaced.

This representation of the Steam Generator losses as 'regulatory' demonstrated the limitations of the coding of events. The data were therefore reassigned into the following three categories: repairs; Tech Spec violations, and inspections. Many of the hours of what had originally been identified as Tech Spec violations and inspections were reclassified as repairs. For example, an inspection outage which lasted more than one month was classified as a repair for the entire length of the outage because steam generator inspections, while lengthy, simply don't take more time than that. Further justification for this reclassification comes from the written descriptions of each event, which stated the purpose of the outages in most cases to be plugging or general repairs.

MAJOR ISSUES OF REGULATORY CAPACITY LOSS
BY OPEC-2 SYSTEM CATEGORY
(in percent)

Issue	Containment	Reactor Coolant	Steam Generator	Undef.	Other	Total
Torus Mods	1.07					1.07
Seismic Bulletins	0.78				0.08	0.86
General Seismic	0.46				0.11	0.57
Snubber Inspections	0.12					0.12
Integrated Leak Tests	0.20					0.20
Seismic Code	0.13	0.08			0.09	0.30
IGSCC		1.09			0.15	1.24
RCS Inspections		0.23				0.23
RCS Tech Spec		0.14				0.14
Steam Gen. Repairs			0.87			0.87
Steam Gen. Inspections			0.43			0.43
Steam Gen. Tech Spec			0.43			0.43
TMI 1 & 2				1.77		1.77
TMI Mods (Non-TMI plants)				0.22	0.02	0.24
Feedwater Cracks					0.19	0.19
Other	0.22	0.10	<0.01	0.05	___	0.37
Total	2.96	1.65	1.73	2.05	0.64	9.03

All other OPEC-2 systems = 1.65
Total Regulatory Losses = 10.68

Table 3.5

MAJOR ISSUES CAUSING REGULATORY CAPACITY LOSS 1975-1984 (in percent)

	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	TOT*
Stress Cracking (IGSCC)	-	0.108	-	1.291	0.438	-	0.038	0.200	2.809	5.648	1.235
Torus Modifications	-	-	-	-	0.165	2.585	2.558	2.379	1.871	0.202	1.069
Steam Gen Repair	-	-	-	0.187	0.158	1.127	1.954	2.454	0.943	1.005	0.872
Seismic Bulletins	-	-	-	-	3.132	4.414	0.047	0.500	-	0.222	0.863
General Seismic	-	-	-	1.020	0.139	-	-	1.230	1.425	1.210	0.568
Steam Gen Tech Spec	0.012	0.163	0.040	0.467	0.364	-	0.610	1.035	0.304	0.927	0.433
Steam Gen Inspections	0.623	1.681	0.721	0.203	0.081	0.299	0.069	0.264	0.309	0.430	0.426
Seismic Computer Code	-	-	-	-	2.899	-	-	-	-	-	0.292
TMI Modifications	-	-	-	-	0.959	1.052	0.292	-	-	-	0.237
RCS Inspections	0.712	0.040	0.231	0.052	0.071	0.257	0.001	0.356	0.275	0.405	0.235
Integrated Leak Test	-	0.271	0.065	0.100	0.085	0.131	0.062	0.581	0.160	0.375	0.197
Feedwater Cracks	-	-	-	-	1.750	0.173	-	-	-	-	0.194
RCS Tech Spec Viol.	0.070	0.010	0.028	0.285	0.263	0.043	0.028	0.242	0.105	0.279	0.144
Snubber Inspections	0.111	0.097	0.090	0.191	0.138	-	0.092	0.036	0.063	0.311	0.116
TMI Units 1 & 2	-	-	-	-	1.243	3.187	3.069	2.916	2.850	2.640	1.774
Total (15 issues)	1.528	2.370	1.175	3.796	11.885	13.268	8.820	12.193	12.703	14.204	8.655
Total Regulatory (All issues)	5.106	5.107	2.841	5.529	13.598	14.958	9.833	13.034	14.256	16.307	10.682

*Ten year average weighted by the number of plants in each year.

Table 3.6

3.3 Data Reclassification

Having reclassified the data in the steam generator category, the reclassification was applied to the outages in all systems. Many of the outage hours of inspections in the Reactor Coolant system were determined to be IGSCC events while other RCS inspections were unrelated to IGSCC and were left in that category. Other IGSCC hours were identified in the Core Cooling/Safety Injection System. Outage hours caused by the Inspection and Enforcement Bulletin on the improper seismic analysis computer code (IEB 79-07) were found to appear in three different systems: RCS, Containment, and Core Cooling/Safety Injection System. The results of this reclassification are shown in Tables 3.5 and 3.6, which show a total of fifteen outage causes. These fifteen causes capture five-sixths of all the regulatory outages, the remaining hours are spread through all other systems. Table 3.5 shows the total losses of the fifteen issues and in which systems they occurred, while Table 3.6 shows the time history of when the events occurred and their effect in each year. These significant issues will now be discussed in detail.

3.4 Significant Issues

Several general observations can be made about Table 3.6 before taking up the specific issues. First, only nine of these issues occurred before the end of 1979 and of those nine, six have been the cause of capacity loss in at least nine of the ten study years. These six constitute the on-going losses. Second, four of the issues appeared for three years or less and none of them have recurred.

3.4.1 IGSCC

Intergranular Stress Corrosion Cracking capacity losses first arose in 1976, again in 1977 and from 1981 to date with significant increases in each of the last two study years.

3.4.2 Torus Modifications

The losses associated with the redesign of the Mark I containments began in 1979, were constant for the next three years, and then fell off by 1984.

3.4.3 Steam Generator Repairs

Steam Generator repairs have been a persistent source of capacity loss from 1978 to the present. These repairs are made to steam generators which fail to meet their technical specifications for allowable leakage rates or are no longer efficient due to the number of tubes plugged.

3.4.3 Seismic Bulletins 79-02 and 79-14

These two bulletins, one concerning the calculations made on the design of pipe support base plates and the other on the differences between the actual pipe layout and seismic calculations, first appeared in 1979. They had a large effect on performance in 1980 and then swiftly dwindled in significance.

3.4.5 General Seismic

Several units were shut down for extended periods while site specific seismic design issues between utilities and the NRC were being resolved. In a few cases the outages were quite long.

3.4.6 Steam Generator Technical Specification Violations

This is an on-going issue, and Table 3.6 indicates that these resulting losses have been increasing over the period of the study. The violations are attributed to temperature deviations, excessive leak rates, water chemistry, and other problems.

3.4.7 Steam Generator Inspections

The losses attributed to Steam Generator inspections are one of the on-going regulatory loss issues. In the years from 1975 to 1978, they accounted for over one-third of the regulatory capacity losses. These inspections are required by the technical specifications either directly or by reference to the ASME Code.

3.4.8 Seismic Computer Code

Bulletin 79-07 was issued in April 1979 and directly affected five plants, four of which had immediate outages, and the fifth was being held out of service after a steam generator replacement. Each plant stayed down for roughly half a year.

3.4.9 TMI Modifications

These were the modifications to plants required by the accident at Three Mile Island and were performed from 1979 through 1981. About equal time was spent in 1979 and 1980 with a pronounced decline by 1981. Since then, no more capacity losses have been attributed to this cause.

3.4.10 Reactor Coolant System Inspections

This is one of the on-going issues of capacity loss. The inspections which were clearly related to IGSCC problems have already been subtracted from this category. The inspections are required under the technical specifications either directly or by invoking the ASME Boiler & Pressure Vessel Code. Some of the inspection requirements also appear in the Code of Federal Regulations, Title 10, Part 55.

3.4.11 Integrated Leak Tests

Containment integrity tests appear in the last nine years of the study. They are performed at the end of a prolonged outage to ensure that the containment does not leak and are required by the regulations of 10 CFR 50 Appendix J.

3.4.12 Feedwater Cracks

The NRC issued IEB 79-13 in June 1979 with revisions in August and October after inspections at several PWRs revealed cracks in the feedwater nozzles leading to the steam generators. All operating PWRs were directed to inspect their feedwater systems and report their findings to the NRC. Any plant which discovered violations of the piping design criteria laid out in the ASME Code was to effect repairs. This problem was dealt with almost entirely in 1979 with a little finishing work in 1980 and is one of the one-time problems which has not recurred. [16]

3.4.13 Reactor Coolant Technical Specification Violations

Violations of the technical specifications governing the temperature, pressure, and water chemistry of the reactor coolant systems have occurred in every year of the study and have always been a small part of the losses. These losses have been constant over the period of the study.

3.4.14 Snubber Inspections

This is the last of the on-going regulatory inspection issues, another problem which has never amounted to very much in a particular year but has been a persistent source of loss. Snubbers are the pipe restraints on the large piping systems. The snubber and the pipe are both subject to fatigue and the technical specifications or the ASME Code require that they be inspected periodically.

3.4.15 TMI Units 1 & 2

The two TMI units were shut down in 1979. Unit 2 will not return to service but Unit 1 has operated in 1985 and is returning to service in 1986. The losses to overall capacity factor have been substantial but, as stated previously, have not been the sole cause of the difference in performance between the United States and the FRG.

CHAPTER 4

4.0 Interviews with U.S. Utility and NRC Personnel

A series of interviews was conducted with utility officials and members of the staff of the Nuclear Regulatory Commission. The purpose of these interviews was to seek additional insight into the regulatory issues which have affected performance in the nuclear industry, and the influence that each has had on plant performance. All interviews were conducted during the first three months of 1986, with an agreement of complete confidentiality.

4.1 The Utilities

Interviews were conducted at six of the thirty-seven utilities which operate the large LWRs considered in this study. Those interviewed were chosen to provide a cross-section of size, experience, location, and most important, performance. The performance standard used was capacity factor. The industry as a whole achieved an average capacity factor of 59.0 percent for the ten years of the study. The capacity loss attributed to regulation, as described in Chapter 3, was 10.7 percent. An overview of the performance of all systems for all other causes in addition to regulatory losses is given in Table 4.1. This table also shows the capacity factors for PWRs and BWRs separately. The regulatory losses have been subtracted out of each system and are presented in aggregate in the Regulatory category.

Figure 4.1 shows the distribution of the utilities for both overall capacity loss and regulatory loss. The overall capacity and regulatory losses for each utility are calculated as a weighted

CAPACITY LOSSES IN THE US*
(in percent)

Type of plant:	All	PWR	BWR
Number of plants:	77	52	25
Plant-years:	622	407	215
Loss Category			
Fuel	1.3%	0.4%	2.9%
Reactor Coolant Sys.	3.9	3.5	4.7
Steam Generators	1.6	2.5	0.0
Condenser	0.6	0.5	0.8
Condensate/Feedwater	1.4	1.4	1.5
Turbine	2.0	2.0	2.1
Generator	1.1	1.3	0.5
Electrical Systems	0.6	0.6	0.6
Refueling	11.2	11.9	9.9
Thermal Eff. Losses	1.7	1.9	1.4
Core Cooling/SIS	0.5	0.4	0.7
Fuel & Grid Economic	1.4	1.0	2.2
Regulatory	10.7	10.5	11.0
Other	3.0	2.4	4.1
Total Loss	41.0	40.2	42.6
Capacity	59.0	59.8	57.4

*Regulatory losses subtracted from each system and totalled in a separate category.

Table 4.1

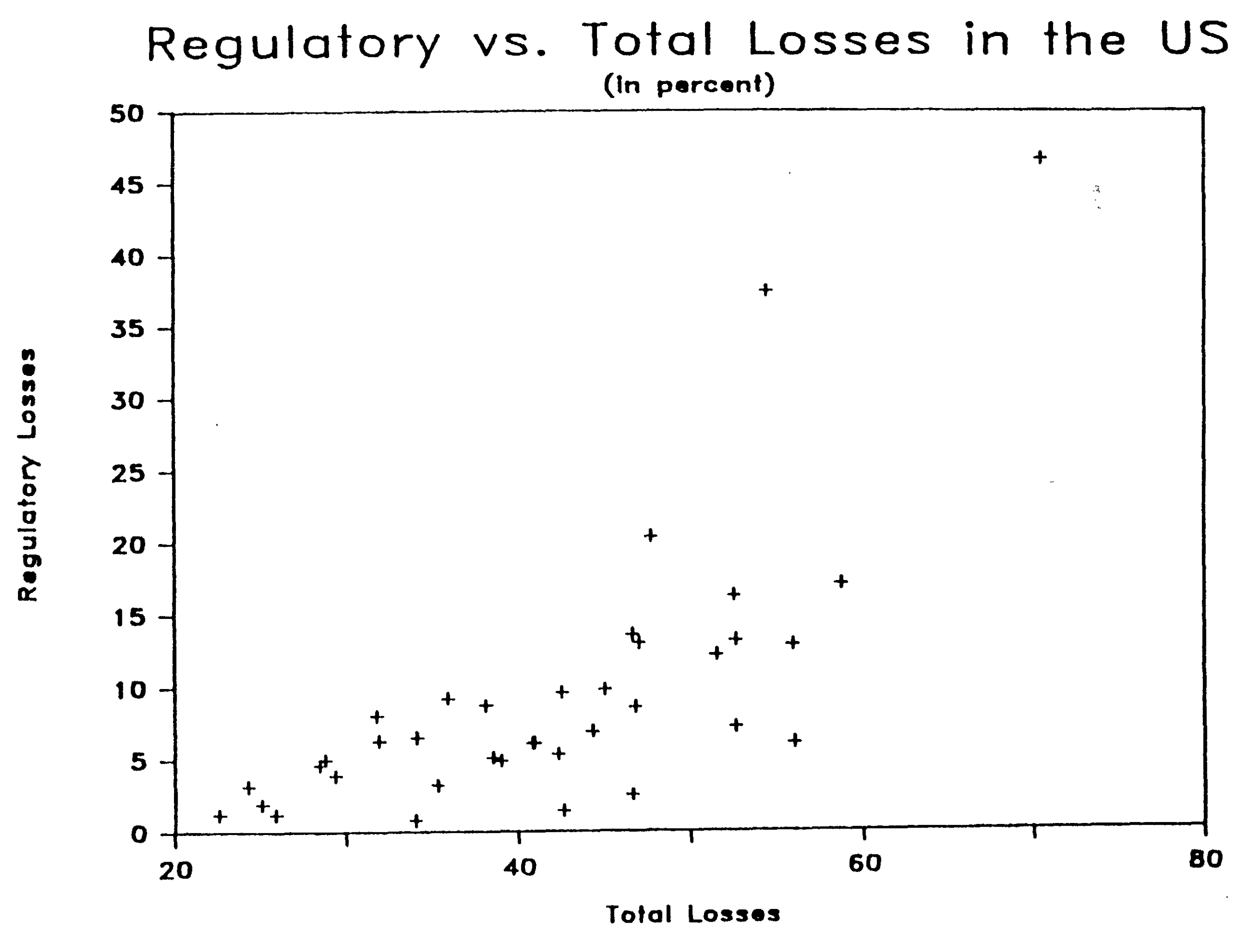
Regulatory vs. Total Losses in the US
(In percent)
Regulatory Losses
Total Losses
0
5
10
15
20
25
30
35
40
45
50
20
40
60
80

Figure 4.1

average of the years of service for each plant. Thus, a plant which only began operations in 1979 has only half as much effect upon the utility's overall performance figures as a plant which has operated for all ten years of the study. A weighted average was used so that no utility would look especially good or bad because of the performance of a single young plant. In this way, Figure 4.1 corresponds to Table 4.1 which uses total hours lost divided by total plant hours available to calculate the capacity factor. The plant hours are the sum of all hours in the year from the date that each plant went into commercial operation until the end of the study period. The figure does not indicate which utilities have many plant-years of experience and does not, by agreement with INPO, identify the individual utilities. It is provided only to show the range of performance in the industry during the ten years of this study.

Using the information of Figure 4.1, a group of utilities was selected for interviews with an eye towards the other criteria mentioned above. One utility owned only one plant, two utilities owned two plants, and three of the interviewees operated three or more plants. The interviews explored the organizational structure of the nuclear operations within the utility, the organization and operation of the safety review process, perceptions of federal and state regulations, and the specific issues identified as regulatory and described in Chapter 3.

4.1.1 Nuclear Operations within the Utility

The utilities have generally separated the operations associated with nuclear power plants from their other power sources. The nuclear organizations are headed by a senior level vice president with access to the utility's chief executive officer (CEO). Each organization is divided into two or three areas: Nuclear Operations, Nuclear Engineering, and perhaps Nuclear Construction. The final area was found only in the utilities with on-going projects; in some cases it was a sub-group of Nuclear Engineering while in others it was a separate division reporting directly to the CEO. Nuclear Engineering and Nuclear Operations were, in a few instances, directed by vice presidents who then reported to a senior vice president.

Nuclear Operations runs the plants. The staff of this division includes the operators, health physicists, chemists, security, and other plant personnel. Their responsibility is for the day to day operation of the plant and, in most cases, they perform little or no engineering. They may oversee repair and replacement work dictated by Nuclear Engineering, but only so far as it affects operations. The size of the staffs vary from 200 to 400 per plant with some of the multi-unit stations able to share such disciplines as security, chemistry, and health physics. Operators are not shared except in the case of twinned units, and then the operator must hold licenses for both units.

The Nuclear Engineering divisions have responsiblity for licensing, quality assurance, and engineering analysis. At some of the utilities they do the actual engineering and procurement for repair and replacement while others act only as project managers for the AE or NSSS vendor doing the work. The size of this division varied widely depending upon the number of plants in operation and, if construction was not a separate division, the number of plants under construction. It ranged from a low of thirty to forty to a high of 500 for a utility with several plants operating and one under construction.

4.1.2 Safety Review

All utilities had an on-site safety review process in which the directors of all disciplines reviewed operating experience and decided upon proposed changes in operating practices, expecially if a change in the plant's Tech Spec or other licensing document would be required. In most cases, this function is performed by a committee which meets an average of once a week and is chaired by the plant or station manager.

Each utility also has an off-site review committee which meets from one to several times a year. The composition of this committee differed significantly among the interviewees. Some draw their members only from within the company, others only from outside, and a few are mixed. On all the committees, several members have extensive nuclear power experience and a few members are drawn from non-nuclear fields. All the members from the utilities are corporate officers and those from outside hold similar rank within their own organizations.

These off-site committees review major changes to the operating license and long term plans for capital expenditures. They may also engage in the systematic review of procedures and practices at the plants and recommend changes in policy. These committees, by being sufficiently distant from daily operations, can provide the utilities with a global assessment of plant performance.

4.1.3 The Utilities and Their Regulators

4.1.3.1 The NRC

The utilities generally described their relations with the NRC staff in Bethesda as good and most got along well with their respective Regional offices. The relations with the Regionals differ according to the particular office, with some of the Regionals described as paternalistic and others adversarial. There exist instances of strong differences of technical opinion between the Regionals and the utilities. One utility said that fewer and fewer of the Resident Inspectors have commercial power experience, most coming from the Navy's nuclear power program or straight out of college, and have little knowledge of operations. The utilities favored the recent reorganization of the NRC staff according to NSSS groups, although some cautioned that it was still too early to tell whether the changes would be effective. They all felt that the old structure had made it too easy for a single technical branch to affect operations, stating that these groups could insist upon work without regard for its effect upon overall plant safety or cost.

A frequently cited example of this is the work required under Appendix R on the separation of redundant components to prevent simultaneous damage from fire. The utilities uniformly felt that the regulations were unnecessarily restrictive and the timetables unreasonably severe. Most of the work has had little effect on operations but has been quite costly.

The utilities readily acknowledged that some of the regulatory issues were significant technical problems, but the timetables for resolution of these problems drew much criticism. The utilities felt that they were forced into faster schedules than were needed from a safety perspective. The result of this was that some of the issues either caused extended shutdowns when the work could have been performed over several outages without affecting operations. A more generous schedule might also have provided the time for more detailed pre-eingineering which would have led to lower costs. While this complaint was leveled by nearly all the interviewees, two provided a contrasting example of how they had performed backfit work over a five year period with NRC approval while the rest of the industry had been forced to make corrections within two years.

A few utilities commented on some of the new tools being used by the NRC, specifically Probabilistic Risk Assessment (PRA) and the Integrated Schedule Program (ISP). The use of PRA was favored by the interviewees. They felt that it was an effective way to set priorities for backfits. One utility observed, however, that a PRA must exist for each plant to properly evaluate the likelihood and danger of a problem. Towards that end they treated the PRA of each of their plants as a "living document", constantly updating it as the plant changes. They emphasized that a one time PRA is ineffective, and that it will carry no weight with the NRC. They also noted that many utilities are openly opposed to developing PRAs for their plants. The reasons given include: the initial cost of performing a PRA; a lack of technical capability within the utility; and a lack of faith in the theory behind PRA. While none of the interviewees professed a disbelief in PRA, some of them had not performed PRAs on their plants and had no plans to do so.

The Integrated Schedule Program (ISP) drew mixed comments from the interviewees. The ISP is intended as a five year plan of modifications, updated each year, showing the schedule of work planned, the budget, and a discussion of priorities for the work. The ISP is then submitted to the NRC for approval. Proponents of the ISP feel that this will reduce regulatory uncertainty and shorten outages. One opponent has stated that he does not want to be locked into a timetable of work and be subject to criticism if other work turns out to be more pressing. Another objection heard from was the assertion that the ISP implies that <u>every</u> issue must be attended to and the only question is when it will be done. To date, only two utilities have adopted ISPs and several others have submitted them to the NRC.

4.1.3.2 The PUCs

The Public Utility Commissions (PUCs) have had little effect on operating performance but several utilities expressed the concern that this would not be the case in the future. The PUCs can become involved with operations through two avenues, fuel cost adjustment hearings and rate cases reviewing capital expenditures.

Most states enacted fuel cost adjustment regulations during the last decade to allow utilities, revenues to keep pace with the rapid increases in oil prices. From 1981 through 1985 oil prices were stable and fuel cost increases have tended to occur when nuclear plants were out of service and power had to be purchased elsewhere. The PUCs have begun setting performance standards for the utilities with penalties when capacity factors fall below a predetermined level. One concern was the selection of one operating cycle as the measure of performance rather than the four year average applied to fossil plants. As a result, a plant might be unduly penalized for a problem which arose in a specific cycle, and the operators might be disinclined to perform preventive maintenance because of the harm to the capacity factor of one measurement period.

Some PUCs have begun to place limits on any capital expenditures made by the utilities on their plants. Expenditures above this limit may not be undertaken without the approval of the PUC. The intention is to prevent unnecessary work from being performed and then added to the rate base. The concern of the utilities is that some of the PUCs lack the technical capability to review this work and that, even when they do not, the reviews can cause great delays in needed work. Moreover, a PUC may be trying to manage operations and thus go beyond its expertise and perhaps its authority. Other utilities have found PUC staffs second guessing technical judgements and, although this has had no effect yet, it may prevent utilities from taking long term action early.

4.1.4 The Issues

The interviewees were affected in varying degrees by the technical regulatory issues detailed in Section 3.3. This section deals with comments on these issues.

Some interviewees lost capacity for repair of damage caused by intergranular stress corrosion cracking (IGSCC). Responses to the problem have covered the spectrum of solutions: full pipe replacement, weld overlay, induction heating stess improvement, and hydrogen water chemistry. There was general agreement that this issue was a problem which required repairs. The question of timing was raised as one utility noted that the NRC appears to be accepting the leak-before-break concept now, and thus those utilities which have not performed full pipe replacement may be able to avoid this costly solution by implementing the other techniques.[1]

The utilities which were affected by IGSCC were also hurt by the problems discovered in the design of the torus of the Mark I BWR containment. On this issue there has not been the broad range of solutions available for IGSCC. The utilities all agreed that the work was required but their actions differed. One utility had prepared an action plan in advance and was able to complete the repairs over the course of several refuelings without extensive capacity loss. Another utility had not prepared a response and was compelled by the NRC to shut down and effect repairs.

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[1]Leak-before-break states that the pipe is made of a material strong enough that even if a through-wall crack develops, it will leak for a long period before it breaks and this leak will be detected by ordinary monitoring systems. The leaking pipe will be identified and the operators would then have ample time to repair or replace the pipe before it broke.

The seismic bulletins of 1979 (IEB 79-14 and 79-02) drew criticism from most utilities and only faint praise from the others. Many doubted the need for this work. The consensus was that these bulletins, while rooted in specific problems, stemmed from a more general desire on the part of the NRC to use new analytical techniques. Also, a belief that the piping systems should be more rigid and required more pipe restraints, or snubbers, was held throughout the industry at that time. A few of the utilities were able to schedule the inspections and subsequent installations of pipe supports in the shadow of outages but others felt compelled to shut down and perform all work immediately. The only favorable comment was oblique. A utility didn't accept the need for the work, but was able to schedule the work during other outages, and benefitted from preparing the as-built drawings. It now has a better description of the plant, and this will help in engineering future work.

The analysis of Section 3.3 divided steam generator losses into repairs (Section 3.3.3), Tech Spec violations (Section 3.3.6), and inspections (Section 3.3.7). The interviewees indicated that the Tech Spec violations were generally leaks which led to inspections and sometimes repairs. Concerning the repairs, all the owners acknowledged that in the absence of regulation, the work would have been performed anyway. The utilities stated that most inspections performed during refueling outages would have been performed whether or not regulations existed and therefore it was unnecessary to call this problem regulatory. One utility did note, however, that it had had to shutdown for a mid-cycle inspection and felt this was appropriately designated a regulatory loss.

One of the utilities interviewed expressed the view that shutdowns by the seismic computer code bulletin (IEB 79-07) were an over-reaction by the NRC. The likelihood that any of the supports had actually been designed incorrectly coupled with the small chance of a severe earthquake while the code was being reviewed was extremely small.

Inspection of the reactor coolant sysem was the tenth most influential regulatory issue of the analysis of Chapter 3. One utility commented that some of the in-service inspection requirements cost time on the critical path of an outage. They felt that the increased sensitivity of the new ultrasonic inspection equipment has begun to detect imperfections which have always been in the welds before but couldn't be seen. They felt that this increased detection rate was unnecessarily raising the incidence of required repairs.

One utility commented on the repairs required on feedwater nozzles following IEB 79-13. They stated that this was a safety problem but felt that the timetable was burdensome. They felt the work could have been performed over a longer period with no increase in public risk and at a far lower cost in dollars and capacity loss. This utility made the same comments on the torus modifications.

4.2 The NRC

4.2.1 Regulatory Tools

The NRC has several tools for shutting down a plant but the staff pointed out that they more often keep a plant from coming back up than shutting it down in the first place.

Among the shutdown tools are : Tech Specs; Inspection and Enforcement Bulletins (IEBs); 10 CRF 50.54f letters; confirmatory action letters from the regional offices; and show cause orders such as the order to all Babcock and Wilcox reactors after the accident at TMI.

After TMI, the number of IEBs issued per year ballooned but has recently declined to a rate of only a few per year. The NRC said that for a period they were regulating by bulletin, a practice the staff described as undesirable. Now there are fewer bulletins issued but more Information Notices which require no response from the utilities.

4.2.2 Changes in Regulatory Practices

The NRC has added four innovations to its regulatory practices since the beginning of the 1980s: the Systematic Analysis of Licensing Performance (SALP); the use of Probabilistic Risk Assessment (PRA) to determine the emphasis which a technical issue receives; the Integrated Schedules Program (ISP); and a staff reorganization along NSSS vendor lines. The SALP has been in use for about five years, PRA explicitly for about two years (although in one form or another since 1974), the ISP has been available to the utilities for about three years, and the reorganization only occurred in 1985.

As the name implies, the Systematic Analysis is being used to evaluate all utilities on their licensing performance. The utility is rated on its ability to meet licensing requirements and good performance will reduce the frequency of future inspections.

Conversely, a low rating will lead to more NRC inspections. The staff noted that more inspections mean that there are more chances of finding something wrong and that it may be hard to get rid of a bad rating without real effort, but they see nothing wrong with this.

PRA is being used to give appropriate emphasis to technical issues. This is used to rank the importance of various problems which arise. The NRC now has an in-house capability to perform PRA, although it still uses an occasional outside consultant.

Although only two plants have ISPs at present, the staff noted that perhaps twenty to thirty other plants have submitted them. The staff acknowledged that many utilities have decided against using ISPs because they don't want to lock their maintenance schedules in with the NRC.

The reorganization has brought together the technical specialists in each discipline and the staff feels that this approach will give them a more systematic view of plant problems. The vendor groups are using PRA extensively when they find a problem.

4.2.3 Selection of Issues

While the NRC is moving toward PRA, the staff stressed that it is still only a tool. If an issue is a serious concern to the staff and they persuade the Commission, the utilities must react regardless of the PRA calculations. The Commission itself also exercises discretion, giving weight to issues which the staff does not necessarily regard as serious. An example of this is environmental qualification of safety equipment (EQ) which became a political issue that the Commission embraced and forced upon the industry without the support of the staff. The staff referred to EQ as a "top-down" issue. EQ has not been studied here because it has had no effect on capacity factors.

The staff also generates issues either from internal research, notification from utilities or vendors, and alerts from other countries. The torus design modification requirements grew from an accident at the West German plant Wurgassen in 1972.

The staff stated that timetables for response to issues are based both upon the severity of the problem and the abilities of the utilities to respond. The severity is generally determined by a PRA. However, many plants do not have PRAs and so they cannot be analyzed. The NRC staff admitted that political pressure may also determine the schedule as in the case of environmental quality (EQ).

4.2.4 The Issues

The staff felt that the flaw in the Mark I Torus design was indisputably a technical problem. They felt that there was footdragging on adopting a schedule for the resolution of this problem and they expressed a similar attitude towards IGSCC.

The staff was divided in its opinion on the requirements for as-built piping drawings in IEB 79-14. Some felt that the absence of precise plant drawings indicated that the plants could not have been properly designed, while others stated that this was a paper chase for some utilities. There was general agreement that this was only a problem for the older plants because the newer ones had better design control and, thus, better drawings.

The staff described the shutdowns required by the seismic computer code bulletin (IEB 79-07) as an over reaction by the

Commission. The problem was detected less than a month after TMI, and the staff asserted that the Commission sought to show that it could move decisively when a problem was detected.

The staff was surprised to discover that the steam generator repairs were described as a regulatory issue. They pointed out that this is a known technical issue and that an economic analysis should dictate good practices. They offered the thought that the steam generator inspections and repairs may be an example of regulations which have enhanced capacity factor.

4.2.5 Discretion of the NRC

The staff acknowledged that there is discretion used in the regulation of the utilities. They felt that the utilities which come to Bethesda and make a good argument for a particular problem solution get their way. Further, they emphasized the discretion of the regional administrators, noting that some of the regions hold paternalistic attitudes while others are adversarial. Commenting on the nature of the entire regulatory process, the NRC concluded the interview by stating that the US as a whole suffers from its adversarial approach.

Chapter 5

5.0 German Data

The German data was compiled from several sources. The VGB, the Association of Large Power Plant Operators, collects and publishes overall performance figures for all German plants each year. [11] "Atomwirtschaft", an FRG power industry journal, publishes graphs showing the monthly operating power of each plant and gives a brief definition of any outage or power reduction which occurred in the past year. [17] The durations of the outages are precise, but the descriptions are quite brief and further clarification was often sought in the publication "Atom & Strom". [18] In cases where the cause of an outage was still unclear, the utilities were consulted.

The FRG data lacks the level of detail of the US data. If a plant was initially shut down for a refueling outage and then remained down to perform other maintenance, the entire outage was described as refueling. On occasions when other work might have been done, there was no description of the critical path time assigned to the non-refueling work. Some of the refueling outages took over four months, and mention will be made here of the other work done, but precise breakdowns of time spent during these outages are unavailable. The performance figures used are energy availability and not capacity factor because the utilities, as part of the long-term agreement to buy coal, at times derate their nuclear plants and instead supply power by running their coal plants. These losses are economic losses and are not related to plant performance; the plants were available to supply the energy but were not permitted to do so because of the obligation to the coal industry.

5.1 Data Analysis

The overall performance of the commercial nuclear reactors of the FRG has been considerably higher than that in the US, although a comparison with Table 4.1 shows that BWRs have performed better in the US. Table 5.1 shows the results for all reactors, for PWRs, and BWRs over the ten year period of the study. The original comparison of US and FRG performance by Hansen and Winje indicated regulatory losses of less than 0.1% in the FRG while Table 5.1 shows a total regulatory loss of 4.4%. The discrepancy arises because of differences between the plant populations samples used in the two studies. The earlier work only examined the 1980-83 performance of PWRs which were 400 MWe or larger, which had been in operation before 1980, and where NSSS and turbine-generator were manufactured by the same vendor. Five German plants and one Swiss plant qualified.

When the BWRs, the newer PWRs, and one small older PWR are added to the database and the study period is expanded from 1975-84, however, the regulatory outage figures increase. The BWRs show over ten times as much regulatory loss as the PWRs. The regulatory outages at all German plants were investigated and will now be discussed.

5.2 Regulatory Losses in the FRG

Only five of the eleven FRG plants identified any regulatory losses. The causes were different in each case and only two of the issues have led to losses of more than a few percent or lasted more than a year. Three PWRs had small losses and two BWRs had the large losses.

ENERGY AVAILABILITY LOSSES IN THE FRG
(in percent)

Type of Plant:	All	PWRs	BWRs
Number of Plants:	11	7	4
Plant-Years:	85	56	29
Loss Category			
Fuel	1.2%	1.4%	0.8%
Reactor Coolant	6.5	0.8	17.9
Steam Generator	0.4	0.5	0.0
Refueling	14.1	15.2	12.0
Turbine	0.6	0.3	1.1
Generator	0.7	1.0	0.1
Condenser	0.5	0.3	0.9
Regulatory	4.4	0.9	11.3
Other	2.4	1.4	4.7
Total Loss	30.9	21.8	48.9
Energy Availability	69.1	78.2	51.1

Table 5.1

In 1976 the staff of Biblis-A discovered cracks in feedwater tanks while inspecting during a refueling. They were given permission to bring the plant back into service with the stipulation that after 1,000 hours of operation (roughly forty-two days) they shut down, reinspect, and perform repairs. The inspection and repairs were completed in four weeks.

From 1980 through 1984, Unterweser was required to reduce power output during several hot weeks in the summer, the losses amounted to 1.0% in 1980, 1.3% in 1981, 4.8% in 1982, 2.9% in 1983, and 0.8% in 1984. The plant's circulating water system draws its cooling water from the Weser river and the temperature of the discharge water is limited for environmental considerations.

The third small loss occurred at Biblis-B and cost the plant 19% of its availability in 1984. A refueling was ordered by the state civil court in Darmstadt because a fuel of greater enrichment than that specified in the license had been installed at the previous refueling. The plant was refueled in two short refuelings and ran at reduced power until all the high enrichment fuel had been replaced.

The two BWRs which had large regulatory losses had very different experiences. Brunsbuttel was held out of service for two years while the Wurgassen plant was only permitted to operate at 80% of full power from 1976 through the end of 1981.

In June of 1978, a main steam nozzle leading into the Brunsbuttel turbine failed. The turbine building filled with reactor coolant and then the plant was shut down,but not before radioactivity had escaped to the atmosphere. The plant, which had only come into service in February of 1977, was shut down to assess and repair the damage. The senior operators were replaced for their handling of the outage which led to the radiation release. The plant was then out for all of 1979 for inspections and training of a new staff. In January 1980 the RSK gave its approval for restart but this was delayed until August 1980 when the BMI finally gave its approval. The BMI delayed the restart while it verified the documentation of the training and inspections. Antinuclear activists then held up the restart by two weeks in the Administrative Court of Schleswig by protesting that an improvement made at the plant affected nuclear safety systems and the license should have been amended. The judge denied the action and the plant finally started and operated for two weeks until a reactor trip from instruments brought the unit down. Meanwhile, a higher court ordered further review of the anti-nuclear objections until the end of October 1980. In November 1980 the plant resumed normal operation although it was down one more time for leaks detected in the reactor coolant system. The six months of outage time in 1978 were attributed to the turbine while the ensuing two years were called regulatory.

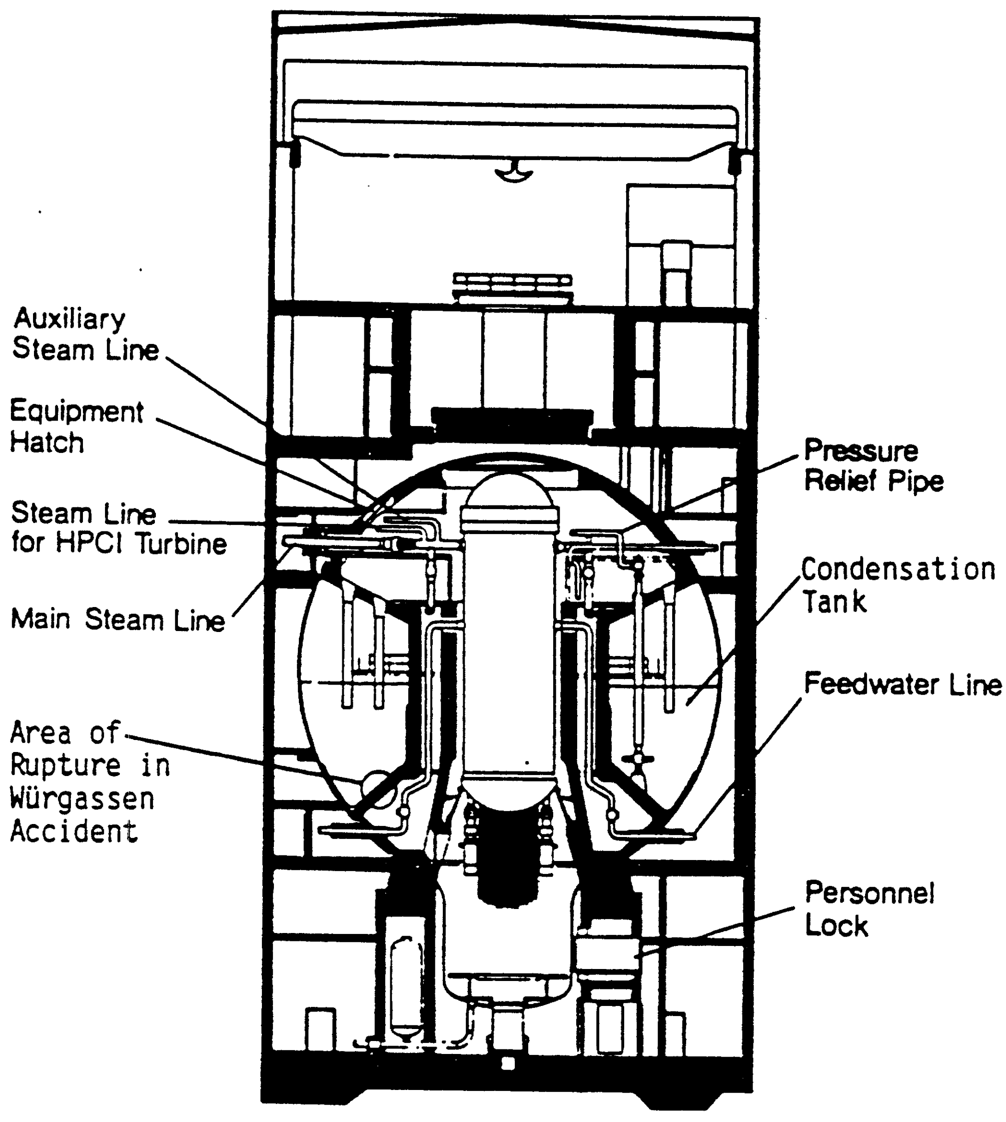
The Wurgassen plant was the most troubled plant in the history of the German industry. Work was completed on Wurgassen by AEG in March 1972 and operational tests then began. In April of that year a pressure relief valve stuck open and admitted steam from the reactor vessel into the condensation tank (the suppression pool or Torus in American BWRs). The valve could not be resealed. Operators tried to shed load instead of going to immediate shutdown but the temperature in the condensation tank rose too quickly. The safety systems worked and the plant was brought down safely, but not before large hydrodynamic loads were experienced inside the containment. These loads burst a wall of the condensation tank which flooded the area below the reactor. A typical layout for a German BWR is shown in Figure 5.1.

The plant stayed down to assess the damage and repair the condensation tank. After the accident, the state regulatory agency in Dusseldorf became concerned about the plant's ability to shut down in another accident. The plant lacked the four fully isolated, independent emergency cooling systems which plants that were receiving construction permits had at that time. It chose to impose an 80% limitation on capacity while these shortcomings existed.

The repairs took a year and then operational tests resumed. A further year elapsed and Preussen-Elektra had still not taken ownership because the plant was not meeting performance standards. A problem then developed in the turbine. Cracks were discovered in both low pressure turbine shafts. New shafts were installed along with new steam dryers and a new steam separator in the reactor vessel to reduce the moisture content of the steam.

Finally, in November 1975, Preussen-Elektra took over ownership of Wurgassen from AEG. The plant operated at 80% of full load from then until it shut down to replace piping and components to bring the plant in line with Basis Safety requirements. At that time diesel generators were added and the core cooling systems were fully isolated and in September 1983 the plant was returned to service at full power. This was the first time that the plant had operated at 100%.

TYPICAL GERMAN BWR LAYOUT



KWU Product-Line 69

Figure 5.1

5.3 Non-Regulatory Losses

As can be seen from Table 5.1, the bulk of the losses came from refueling and maintenance at all plants and the Reactor Coolant System in BWRs. The RCS losses were the result of a revised approach to safety called the Basis Safety Concept, which will be discussed in Section 5.4.

5.3.1 Refueling and Maintenance

This is the only category which would be expected to have consistent losses exceeding ten percent. An investigation of the longer outages, however, established that work going beyond ordinary maintenance was performed during these outages. Three significant issues will be discussed here.

The refueling outages at Biblis-A and Biblis-B in 1980 lasted six months and four and a half months respectively. During those two outages, all the reactor vessel core barrel bolts were replaced. The original bolts, of a higher strength than those used in previous plants, were found to be subject to corrosion embrittlement. The bolts were removed underwater and replaced with bolts of the steel used in older designs.

The refueling outage at Neckarwestheim in 1979 took nearly six months, during which time valves and other components in the emergency core cooling system were replaced. A total of nearly 250 tons of steel was replaced during this outage. This was the only prolonged outage at an operating PWR to upgrade piping to meet the Basis Safety requirements.

The last extended refueling outage occurred in 1983 when the Biblis-B plant was down for three months. The hydrogen cooling system for the generator failed, and although the generator was not seriously damaged the repair work took time. This was not a nuclear problem and could equally well have happened at a fossil plant.

5.3.2 Steam Generators

There have been only two outages of any length related to steam generators in the FRG. Both steam generators at the 350 MWe Obrigheim plant were completely replaced in 1983, the outage lasted two and a half months, and was performed during a three and a half month outage that was actually identified as refueling for the entire length of the outage. Obrigheim is the oldest operating PWR in the FRG, having come into service in March 1969. In 1984, Biblis-A reported a one month outage for repair and testing after detecting primary to secondary side leakage in one of the steam generators. It must be noted that in sharp contrast to the US none of the steam generator outage time has been called regulatory, not even inspections.

The inspection standards used by the FRG utilities are higher than those required by the KTA guidelines. In 1980, the PWR owners and KWU became concerned about the possibility of steam generator problems and began to increase their inspections. At Neckarwestheim, in fact, the utility inspected 100% of the tubes each year year from 1981 through 1983. They adopted this approach in the hope of learning where the corrosion problems would occur and now that they have identified the susceptible areas, they are reducing the inspection levels. A new set of KTA rules governing the steam generator has been proposed which reduce the extent and frequency of inspections to a level equal to that of the US.

5.3.3 Fuel

The two Biblis units ran out of spent fuel storage space in 1979 and applied for license amendments to install high density fuel storage racks in their spent fuel storage pools. The licenses were delayed so long in the courts that the owner, facing a large winter demand, elected to derate the plants rather than have them unavailable. Biblis-A ran at fifty percent power for nearly five months and Biblis-B at fifty percent from July through March of the following year. Only Biblis-B received permission to install the new racks on the grounds that since it was located further from the river and its fuel pool wall was thicker than Biblis-A, it was more capable of sustaining the shock of the explosion of a liquified natural gas (LNG) tanker in the nearby river. The courts accepted this resolution and Biblis-A ran at full power from the beginning of December until its refueling in March at which time Biblis-B returned to full power until its refueling in August of 1980.

5.3.4 Electrical Switchgear

In 1977, Biblis-A was derated by 7% in February when a 380 KV transformer failed. A replacement transformer was not available until June when the plant returned to full power.

5.4 The Basis Safety Concept

In 1972, materials engineers at the University of Stuttgart began to reevaluate the nuclear industry's approach to the design of pressure vessels and high pressure piping systems. Catastrophic failure of these components was viewed as a possibility, albeit slight. At the same time, concern had arisen over the variety of design standards and guidelines in use for these components. The Basis Safety Concept (Basis Sicherheitskonzept) was proposed to bring all the standards under one heading and at the same time raise the standards to such a level that a catastrophic failure would be incredible. [19]

Basis Safety began with materials, but it is also an overall design concept. Members of the industry had begun to recognize that some of the high strength steels being used lacked the flexibility to respond to thermal and pressure shock and also that they became brittle in the areas near welds. To respond to this problem, Basis Safety proposed to use more forgiving material, and to increase the pipe wall thickness; to reduce the number of welds and move those welds which remained away stress points in the piping systems; and finally, to locate the welds where in-service inspections could be done easily.

The Reactor Safety Commission (RSK) adopted Basis Safety in principle in 1977 and in 1979 published its first set of guidelines on the subject. The guidelines, which have been updated several times already, include stringent requirements on: material quality; additional quality assurance by multiple parties independent of the manufacturer; continuous in-service inspection; continuing programs in research and development; and materials testing to verify the validity of the standards. The regulations apply to plants under construction, operating plants are immune to backfit requirements except in cases of "imminent danger."

An event which lent momentum to the Basis Safety initiative was the discovery of cracks in the reactor coolant piping of several

German BWRs which were under construction at the time. The RSK, acting on behalf of the Federal Minister of the Interior (BMI) began an investigation into possible causes and the systems and components affected. This study identified:

- parts of the main feedwater and steam lines;
- buffer tanks of the scram system;
- parts of the preheater and reheater-condensate cooler in the turbine building;
- parts of the branch connections to the pressure-suppression system;
- parts of the auxiliary steam supply system. [20]

The research identified the desirability of using materials which are "more 'forgiving' to deviations from specificed manufacturing parameters and have a high fracture toughness," [20] a clear echo of the Basis Safety policies.

At the end of 1979, three BWRs were operating in the FRG and a fourth was undergoing its first startup tests. These plants were never ordered to shut down, but they received gentle pressure from the RSK with the warning of increased inspection requirements if they didn't replace the pipe. From 1980 through 1983 each of the four plants shut down for periods which lasted from twelve to seventeen months. During this time they replaced all their reactor coolant and feedwater piping as well as main steam isolation valves. Some utilities also took the opportunity to replace the pre-heaters and the economizers and to perform repairs on the reactor vessel steam-water separator and the turbine moisture separator. These were moves to improve long term availability. Over this four year period, the BWRs in the Federal Republic had an aggregate availability of 46.7%.

The only PWR to be affected was the previously mentioned Neckarwestheim. The primary coolant piping of the PWRs was already of a high quality and it, along with many of the major components, was reanalyzed but survived. Secondary side piping was replaced, but all this work was done in the shadow of several refueling outages.

5.5 IGSCC

The only instance of IGSCC in Germany occurred at the 250 MWe Gundremmingen-A plant which was shut down and decommissioned in 1977. This plant is not a part of this study because of its small size. Gundremmigen-A is the only German plant which was built under license to General Electric. The piping used in that plant followed the GE specifications and was the same Type 304 unstabilized austenitic stainless steel used in the US BWRs. The plant was already down for repairs following as accident when the IGSCC was discovered and the extent determined. The decision to decommission was made based on the age of the plant and its small contribution to overall capacity.

The newer German plants use Type 347 stainless steel in all piping less than twelve inches in diameter and clad the inside of the larger pipes with the 347. This steel, which is difficult to weld, is stabilized against sensitization and can't develop IGSCC. The German industry selected this material to eliminate the possibility of IGSCC even though the chances of it occurring seemed remote at the time.

5.6 Seismic Design Standards

The KTA adopted the seismic spectrum and seismic analysis regulations of the AEC/NRC in 1975. While no zone in the FRG is as active as the Pacific Coast of the US, most of the country bears a seismic resemblance to the American East and Midwest. The dynamic analysis regulations of the FRG stipulate that they must design to the greatest load sustained from a seismic event, a chemical explosion (such as an LNG tanker on a river near the plant), or a jet airplane crash. The aircrash, when analyzed, is a more severe loading problem than a seismic event. The aircrash stipulation was added in the early 1970s and affected several plants in this study while they were under construction. There was no effect on operating plants.

The analytical capabilities of the two countries are and have been comparable. Like the US, the FRG went through a period in the late 1970s of intensive reanalysis of the seismic designs of the plants under construction and in design. They investigated every component and now KWU feels it has established a store of knowledge large enough to perform analysis by similarity in some cases.

The analysis has moved the FRG towards a "soft" design which uses almost no snubbers in the new plants. The older plants do have snubbers and they resisted the pressure to add more snubbers in the late 1970s and early 1980s. Their resistance paid off because by 1982 the analysis indicated that fewer snubbers rather than more increased the survivability of the piping systems. When piping is modified at the old plants, some snubbers are removed, but there is no active program to remove the installed snubbers.

The only snubbers that have been added are to prevent damage from pipe whip. During the intense analysis period from 1976 to 1982 KWU discovered that some of the greatest dynamic loads come from pipe breaks and the resulting pipe whip. They have reanalyzed for this problem and the utilities, on their own initiative, have added snubbers.

5.7 In-Service Inspection

The present KTA rules for inspection of the primary system components and piping equal or exceed those of the US. All welds in the reactor vessel and primary piping must be inspected at least once every four years. [21] The inspections are performed by the utilities using remotely operated ultrasonic testing equipment. The TUVs witness the inspections and will actually perform a few tests to verify the quality of the utilities, inspections. A revision of the KTA rules has been proposed, using the ASME code as a guide to reduce the frequency of inspections. The standards for the reactor vessel will remain the same but the period for piping and other components will be extended to eight years. The utilities are encouraging this revision but other members of the KTA are reluctant to make this change; it is unclear when, if ever, this rule will be revised.

5.8 No-Loss Regulatory Issues

There have been a few instances in the FRG of regulatory issues which have led to work at operating plants but had no effect on operations. Like the US, the FRG has added hydrogen recombiner equipment in the aftermath of the accident at TMI. Every plant in the FRG has been required to add a backup secondary side cooling system. This consists of an extra cooling tower, reserve water supply, pumps, and diesel generators. The backup cooling system must be located at a distance from the original cooling system so that they won't be damaged simultaneously by the same airplane crash.

Chapter 6

6.0 The Effects of Regulation

The picture which can be drawn from the comparison of US and FRG data illustrates that most of the difference in regulatory losses in the two countries is a matter of definition. In the FRG, the only issues which have been called regulatory were a derating of an operating plant, an order to train a new staff after an accident, and some environmental operating limits. In the US the regulatory issues encompassed design flaws (IGSCC and torus modifications); inspections of the primary system boundary, either in the steam generators or the reactor coolant system; leaks or potential leaks in the primary system boundary in the form of Technical Specification violations for the steam generators or reactor coolant system; and reanalysis of the seismic design criteria (the seismic bulletins, the seismic computer code, and plant specific seismic issues).

6.1 The Issues Side by Side

Intergranular Stress Corrosion Cracking was a problem in both countries but in Germany was mostly avoided by the selection of Type 347 steel. The one plant that did have the problem never returned to service. There was no question on the part of the US industry that this was a technical problem, and that the piping was eventually going to crack. The only question concerned how extensive the repairs needed to be. Many in the US have argued that increased inspections, weld overlay, and the leak-before-break concept make immediate full-scale pipe replacement unnecessary. Under similar circumstances the German industry rejected this approach. The knowledge that the material chosen for the reactor coolant and feedwater piping might crack was enough to prompt the Basis Safety backfits. The solution may not have been agreeable to all members of the industry, but it was accepted over this option of increased inspections.

The accident at Wurgassen in 1972 showed the German industry the design flaw in the condensation tank and tipped off the US to possible flaws in the Mark I Torus. Even after the repairs were completed at Wurgassen, the plant was held to 80% of full power for many years.[1] The US, after verifying that this could be a problem, studied the problem from 1975 to 1979 before taking action, and many utilities stated that they needed still more time. None of the US plants were ever subject to an 80% power restriction.

The US industry has credited regulation with over half of all the losses suffered by steam generators from 1975 to 1984, while none of the steam generator losses in the FRG have been regulatory. The German industry voluntarily increased its inspections when it saw the trouble with the American steam generators, in the hope of identifying trouble spots and taking preventive action.[2] The locations of corrosion must be identified and measures taken before tube plugging becomes necessary. Once a tube is plugged, the heat

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[1] This regulatory action is, at least, questionable; there is little if any increase in safety in bringing down a plant from only 80% power under accident conditions instead of the full 100%.

[2] This volunteerism should not be confused with the "voluntary" pipe replacement in the BWRs. In the case of steam generator inspections there was no hint of a desire for increased inspections from the regulators.

transfer capacity of the steam generator is reduced and the plant will produce less power. The US and FRG regulations for tube plugging both use a threshold of 50% wall thinning. The FRG regards this as a plant availability problem and some US utilities acknowledged that there was little reason to call this regulatory.

The attitude towards inspections of the primary system boundary is markedly different in the two countries. The FRG inspection standards have equaled or exceeded the US in every category and none of the time spent has been labeled regulatory. As with steam generators, inspections are viewed as a way to maintain high availability. Piping and components have failed, and will continue to do so, even under a policy of Basis Safety, and inspections are the only way to anticipate and prevent these failures. Inspections are performed at fossil as well as nuclear power plants; inspections are a part of normal plant operations.

A reasonable parallel may be drawn between the outages at Brunsbuttel and TMI-1. The American plant, although undamaged, was of a similar design and operated by the same organization as TMI-2. The unit was already shut down for refueling and in the aftermath of the accident, when operator error was shown to have contributed to the problem, the NRC had reason to hold the plant down for inspection and training. The response to the Brunsbuttel accident, which predated TMI by nearly a year, seemed unusually harsh - no permanent damage was done and the total release of radiation was small - but the doubts about the operation team's abilities led to a complete replacement of operators and a two year outage. This response, although draconian, is perhaps more suitable than the seven years of outage at TMI-1.

Of the major regulatory issues, only the seismic reanalysis in the US has no parallel in the FRG. The NRC and the utilities have acknowledged that the shut down order for the seismic computer code problem was an overreaction, which was related to the fact that it occurred less than a month after TMI. (Some U.S. observers suggested that the Commission may have sought to maintain its credibility by showing that it could react decisively to a problem.)

The other seismic bulletins and the plant specific seismic issues lack the political implications of the computer code issue but have accounted for nearly five times as much lost capacity. The NRC went well beyond the German response to the improvements in analytical capabilities. The German regulators' reluctance to demand backfits while the analysis was still being developed served the industry well as the new analysis finally showed that fewer, rather than more, snubbers were desirable. In comparison with the FRG, the US reanalysis and backfit requirements appear to have been precipitous.

The FRG has never challenged the overall seismic design of a plant as has happened at a few US plants. In one instance, when Biblis-B was under construction and using a higher design standard that its sister plant, KWU reviewed the seismic design of Biblis-A anticipating a question from the regulators. The question never came but the plant met the higher design standard anyway.

6.2 Differences in Approach

Both industries suffered losses due to the replacement of main coolant piping. The technical problems were different, the IGSCC in the US had caused the austenitic steel to leak, while in the FRG the high strength ferritic steel was regarded as too brittle. The German industry may refer to the pipe replacement at the four BWRs as voluntary, but there is ample evidence to suggest that more than "gentle pressure" was applied by the regulators in their move to promote Basis Safety. [22] The approach to developing the technical solution to these two problems was quite different and it is worth exploring those differences here.

Basis Safety is not so much revolutionary as it is evolutionary. The change in design standards came when the technology improved: new materials; materials manufacturing techniques; analytical techniques; and inspection abilities. These innovations were then set against the backdrop of experience which showed the need for inspections and the recognition of designs which led to fatigue and possible failure.

In Germany, the response was the formulation of new regulations. In the US, the manufacturers and utilities adopted a similar approach of their own accord. The NRC specified that the cracked pipe be replaced with pipe immune to IGSCC, but they left it to the industry to choose the material, the design, and installation methods. When General Electric offered its bid to utilities planning to replace damaged piping, the design embodied the concepts of Basis Safety. The material was of a higher grade, there were fewer welds, those welds which remained were distant from stress points and in locations where inspections could be done easily, and the welding techniques were improved.

A relevant question to ask in this comparison is of the relative timing of the responses? It is not enough to say that the industry would have arrived at the same point of design without regulation if that point is reached five years later. The redesigned BWR piping in the US was offered at about the same time that the new piping was being installed in the BWRs in the FRG, but this may have happened because the German industry had forced the frontiers and the US industry benefitted from this work. How much work was done in isolation and how much together is unclear.

What is clear, however, is that the original requirements for inspection, leak detection, and allowable leakage rates were the cause of the enhanced design in both countries. The inspection and leak detection requirements provided the alert to design deficiencies, and the leakage limits provided the economic incentive to replace the piping. To avoid the requirement to shut down for frequent inspections and repairs, it became advantageous to install materials of higher quality and lower failure probability.

6.3 Lessons Learned

The FRG has had more capacity loss due to regulation than it has admitted, and the US less. The total for the FRG agruably should include the losses for pipe replacement under Basis Safety (although the possibility that some pipe replacement losses would have been incurred even in the absence of the Basis Safety initiative must also be recognized). This would add 5.4% for the four BWRs, and 0.5% for the PWR Neckarwestheim. The US should arguably remove steam generator repairs (0.9%) and perhaps the steam generator inspections and Tech Spec violations (0.4% each) on the grounds that steam generators need to be inspected because they suffer from corrosion which causes leaks, and when there is a chance of a leak, corrective action must be taken. The same arguments can be applied to inspections and Tech Spec violations of the reactor coolant system (0.2% and 0.1% respectively). The result of all these additions and subtractions is shown in Table 6.1, which suggests the surprising conclusion that regulatory losses have actually been greater in the FRG than the US. It follows that regulation is not the cause of the poorer performance of the US nuclear industry with respect to that of the FRG.

It must be noted here that not all US utilities have had great losses attributed to regulation, and also that some have operated on a par with the FRG plants. The key similarities between US and FRG utilities detected during the interviews are:

(a) inspection levels above the regulatory standards;

(b) acknowledgement of problems before the regulatory authority becomes concerned;

(c) development of resolutions in anticipation of or to forestall regulatory action; and

(d) a willingness and an ability to spend money to fix a problem.

Based on the data and the interviews, it is conjectured that those utilities which performed inspections above and beyond the requirements were the utilities which performed well overall. These

RECALCULATED REGULATORY LOSSES
(in percent)

	US	FRG
As Stated	10.7%	4.4%
Basis Safety Backfits		5.9
Steam Generator Repairs	(0.9)	
Steam Generator Inspections	(0.4)	
Steam Generator Tech Spec	(0.4)	
RCS Inspections	(0.2)	
RCS Tech Spec	(0.1)	
Total Losses	8.7	10.3

Table 6.1

utilities may have been able to identify problems long before they became industry wide issues, whereas those which inspected reluctantly and only in accordance with regulations were likely to be less prepared and may have lost significant amounts of capacity when industry problems were discovered. The latter may also have had problems because they had done fewer inspections and didn't know their plants and their shortcomings as well, although this is not clear from the investigation.

Caution must be exercised here. This should in no way be construed as a call for more severe inspection requirements. Inspection regulations must set a minimum threshold for safety. Plant availability is the responsibility of the utilities and they must decide how to allocate their resources to make the plants perform well. A concern expressed by utilities in both countries is that if they follow higher inspection standards, the regulators will require that those standards be adopted permanently. This must not happen. The utilities must have the flexibility to investigate an issue for a few years and when they have concluded their investigation, turn their attention to other questions, as the Germans have done recently with steam generators.

The utilities which have been willing to acknowledge a problem are the ones which can begin working on resolutions. It has not been easy for some utilites to acknowledge a problem because of the political nature of nuclear power. This has been a particular problem for utilities with ongoing construction. Every new problem detected at an operating plant tends to raise new questions about the decision to invest in a new plant. This tends to promote a wait-and-see attitude which allows problems to grow, partly in the hope that other plants will also develop the problem and an industry wide consensus will develop on the need for work. This wait-and-see attitude is not, however, peculiar to utilities with new construction. Those with no new construction work and only operating plants often want to identify all the other victims of a problem when they apply to their PUC for compensation for money spent. This approach, unfortunately, again allows a problem to grow before a solution is tried.

The data and the interviews suggest that the NRC has been and is growing more receptive to proposals to extend implementation schedules presented by utilities which show a detailed understanding of the problem and its resolution. The utilites which have presented detailed repair plants to the NRC on seismic backfits and torus modifications are the ones that have been able to stretch out their repairs over several years with little impact on operations. Those utilities which have used this approach successfully have usually offered to increase on-line surveillance over the course of the repair period.

6.4 Future Work

This thesis has concluded that regulation is not the cause of poor U.S. nuclear power performance relative to the FRG. It has suggested that the investment in inspections, early recognition of problems, and development of solutions in advance of regulatory action are all keys to good performance. Future work that could shed further light on the latter issue would consist of an investigation of inspection practices in the U.S. and abroad to try to correlate them with performance.

Another factor that was not specifically addressed in the present study but which may explain some of the differences in performance and which deserves further analysis is the effect of differences in the financial environment of different utilities. For example, German utilities are generally both able and willing to spend money on plant improvements. However, not all US utilities are willing to spend money even if they are able to do so; those with ongoing construction projects may be deterred from making changes in operating plants because of the risk of prejudicing public perceptions concerning the wisdom of the new investments. Moreover, not all US utilites which are willing to spend money on plant improvements are able to do so. The PUCs have become increasingly reluctant to allow the utilities to spend money. They are demanding ample justification of all expenditures, much of this in the form of examples of other utilities which have already done the same work. This tends to discourage early preventive maintenance.

Another problem which may prevent some U.S. utilities from spending money when they need it is the so-called cost cap. A few PUCs have already established and others are considering a ceiling on plant expenditures above which the utilities must seek the approval of the PUC to perform the work. Not only does this delay the work but it may also call upon the PUC to make technical assessments that it is unqualified to make. This is not to say that the PUC should abrogate its right to review expenditures, but it should not command that the utilities give up their responsibility to make decisions for their plants. If the PUCs want to ensure that the expenditures are reasonable, they may want to consider setting penalties for low performance, but to make these disincentives effective work they should be coupled with incentives for high performance.

Differences in financial influences and incentives at the construction stage may also be important. Nuclear power plants are built in the FRG in roughly seven or eight years. The utilities pay for the plants with accumulated revenues, loans, and, if they choose, current revenues. They are allowed to incorporate part of the interest on the loans in the electric rates prior to completion of construction (usually referred to as CWIP in the US). The interest and inflation rates in the FRG have both been lower than in the US over the past twenty years.

The electric utility industry in the FRG is obligated to purchase coal from the German coal industry. This domestic coal is expensive and drives up the cost of electricity. In comparison, nuclear power is economically attractive.

The Price Commission in each state has the right to review all expenditures before allowing them into the rate base. However, the Price Commissions have shown greater interest in making sure that a utility's costs are consistent with other German utilities than determining whether the costs were the lowest attainable. The only cost review that is done comes when the project is first proposed. If the costs are in line with other projects in the FRG, the utility never has to submit another cost analysis for approval by the regulators.

In the US, by contrast, electricity from coal-fired plants is not as expensive as in Germany. Nuclear plants have taken an average of ten to twelve years to build, CWIP has been allowed in only a few cases, and the utilities are subject to ongoing cost-justification reviews during construction and later during operations. Under these pressures, the US utilities find themselves forced to give great consideration to construction costs. The question is whether, as a result of these differences, the German utilities have tended to invest in higher cost, but also quality systems during the construction phase.

Recommendations for future work that focus on these financial and economic issues include:

o A comparison of the quality of components, both major and minor, installed in plants in both countries:

A comparison of designs for reliability and maintainability.

o A comparative investigation of economic incentives for good performance.

o Comparative research on economic incentives to build in high quality from the outset, including the possibility of performance guarantees in future contracts.

References

1. Hansen, K. F., D. K. Winje, "Disparities in Nuclear Power Plant Performance in the United States and the Federal Republic of Germany", Massachusetts Institute of Technology Energy Lab, MIT-EL 84-018, December 1984.

2. U.S. Energy Information Administration, Energy Facts 1984, May 1985, Washington, D.C. and the INPO database.

3. "Operating Plant Evalutation Code - 2", prepared by the S. M. Stoller Corporation, Boulder, CO.

4. U.S. Code, Title 42, Section 5846(d).

5. Anderson, D. D. "State Regulation of Electric Utilities", in James Q. Wilson, ed. The Politics of Regulation.

6. "Die Öffentliche Elektrizitätsversorgung" 1982 and 1983, Vereingung Deutscher Elektizitätwerke, Frankfurt.

7. Nuclear News, "World List of Nuclear Power Plants", Nuclear News, August 1985.

8. "Elektizitätswirtschaft", 1985, Vereingung Deutscher Elektizitätwerke, Frankfurt.

9. Act on the Peacful Utilization of Nuclear Energy and the Protection against its Hazards (Atomic Energy Act) (Gesetz über die friedliche Verwendun der Kernergie und den Schutz gegen ihre Gefahren (Atomgesetz) of December 23, 1959.

10. Franzen, L. F. The Nuclear Licensing and Supervisory Procedures for Nuclear Facilities in the Federal Republic of Germany. Published by Gesellschaft für Reaktorsicherheit mbH, Cologne, W. Germany, 1982.

11. Annual Report 1983/1984, VGB Technische Vereinigung der Grosskraftwerksbetreiber, and all other annual reports from 1975 through 1984.

12. Koppe R. H., et. al., Nuclear Unit Operating Experience: 1980 Through 1982 Update, Prepared by the S. M. Stoller Corporation, Boulder, CO for the Electric Power Reasearh Institute, EPRI publication No. NP-3480, April 1984, Palo Alto, CA.

13. U.S. N.R.C., IE Bulletin No. 79-02 Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts, and revisions. First issued March 5, 1979.

14. U.S. N.R.C., IE Bulletin No. 79-14, Seismic Analyses for As-Built Safety-Related Piping Systems, and revisions, first issued July 2, 1986.

15. U.S. N.R.C., IE Bulletin 79-07, Seismic Stress Analysis of Safety-Related Piping. Issued April 14, 1979.

16. U.S. N.R.C., IE Bulletin 79-13, Cracking in Feedwater System Piping, and revisions, first issued June 25, 1979.

17. "Atomwirtschaft", published by Vereingung der Grosskraftwersbetreiber, 1975-1984.

18. "Atom & Strom," published by Verlags-und Wirtschaftsgeselshaft der Elektrizitätwerke mbH, Frankfurt, 1975-1984.

19. Kussmaul, K., "German Basis Safety Concept Rules Out Possibility of Catastrophic Failure", published in Nuclear Engineering International, December, 1984.

20. Schulz H., Mueller W., "Piping and Component Replacement in BWR System; Safety Assessment and Licensing Decisions," Nuclear Engineering and Design 85 (1985), pp. 177-185, Elsevier Science Publishers B. V., North-Holland, Amsterdam.

21. KTA Regulation 3201.4, "Light Water Reactor Primary Circuit Components: Periodic Inspections and Monitoring" ("Komponenten des Primärkreises von Leichtwasserreaktoren Teil: Wiederkerhrende Prüfungen und Betriebsüberwachung"), Kerntechnischer Ausschuß, issued June 1982.

22. Kussmaul K., Blind, D., "Basis Safety - A Challenge to Nuclear Technology", Proceedings of IAEA Specialist Meeting on Trends in Reactor Pressure Vessel and Circuit Development, Madrid, March 5-8, 1979.

APPENDIX 1
U. S. PLANTS USED IN THIS STUDY

PLANT NAME	NET RATING	TYPE	COMMERCIAL:
ARKANSAS 1	820 MWe	PWR	1/75
ARKANSAS 2	912 MWe	PWR	4/80
BEAVER VALLEY 1	852 MWe	PWR	3/77
BROWNS FERRY 1	1065 MWe	BWR	8/74
BROWNS FERRY 2	1065 MWe	BWR	3/75
BROWNS FERRY 3	1065 MWe	BWR	3/77
BRUNSWICK 1	821 MWe	BWR	4/77
BRUNSWICK 2	821 MWe	BWR	12/75
CALVERT CLIFFS 1	880 MWe	PWR	6/75
CALVERT CLIFFS 2	880 MWe	PWR	4/77
CONN YANKEE--HADDAM	582 MWe	PWR	1/68
COOK 1	1054 MWe	PWR	9/75
COOK 2	1100 MWe	PWR	7/78
COOPER STATION	778 MWe	BWR	7/74
CRYSTAL RIVER 3	856 MWe	PWR	4/77
DAVIS-BESSE 1	906 MWe	PWR	4/78
DRESDEN 2	794 MWe	BWR	7/72
DRESDEN 3	794 MWe	BWR	1/72
DUANE ARNOLD	515 MWe	BWR	2/75
FARLEY 1	829 MWe	PWR	12/77
FARLEY 2	829 MWe	PWR	8/81

PLANT NAME	NET RATING	TYPE	COMMERCIAL:
FITZPATRICK	821 MWe	BWR	8/75
FORT CALHOUN	478 MWe	PWR	7/74
GINNA	490 MWe	PWR	4/70
HATCH 1	786 MWe	BWR	1/76
HATCH 2	784 MWe	BWR	10/79
INDIAN POINT 2	873 MWe	PWR	7/74
INDIAN POINT 3	965 MWe	PWR	9/76
KEWAUNEE 1	535 MWe	PWR	7/74
LASALLE 1	1078 MWe	BWR	1/84
LASALLE 2	1078 MWe	BWR	11/84
MAINE YANKEE	825 MWe	PWR	1/73
MCGUIRE 1	1180 MWe	PWR	12/81
MCGUIRE 2	1180 MWe	PWR	3/84
MILLSTONE POINT 1	660 MWe	BWR	4/71
MILLSTONE POINT 2	870 MWe	PWR	1/76
MONTICELLO	545 MWe	BWR	8/71
NINE MILE POINT	610 MWe	BWR	1/70
NORTH ANNA 1	907 MWe	PWR	7/78
NORTH ANNA 2	907 MWe	PWR	1/81
OCONEE 1	887 MWe	PWR	8/73
OCONEE 2	887 MWe	PWR	10/74
OCONEE 3	887 MWe	PWR	1/75
OYSTER CREEK	650 MWe	BWR	1/70

PLANT NAME	NET RATING	TYPE	COMMERCIAL:
PALISADES	740 MWe	PWR	1/72
PEACH BOTTOM 2	1065 MWe	BWR	8/74
PEACH BOTTOM 3	1065 MWe	BWR	1/75
PILGRIM 1	668 MWe	BWR	1/73
POINT BEACH 1	497 MWe	PWR	1/71
POINT BEACH 2	497 MWe	PWR	5/73
PRAIRIE ISLAND 1	530 MWe	PWR	1/74
PRAIRIE ISLAND 2	530 MWe	PWR	1/75
QUAD CITIES 1	789 MWe	BWR	3/73
QUAD CITIES 2	789 MWe	BWR	4/73
RANCHO SECO	917 MWe	PWR	5/75
ROBINSON 2	730 MWe	PWR	4/71
SALEM 1	1090 MWe	PWR	7/77
SALEM 2	1115 MWe	PWR	11/81
SAN ONOFRE 1	430 MWe	PWR	1/68
SAN ONOFRE 2	1087 MWe	PWR	9/83
SAN ONOFRE 3	1087 MWe	PWR	4/84
SEQUOYAH 1	1148 MWe	PWR	7/81
SEQUOYAH 2	1148 MWe	PWR	6/82
ST. LUCIE 1	846 MWe	PWR	1/77
ST. LUCIE 2	804 MWe	PWR	9/83
SUMMER 1	900 MWe	PWR	1/84
SURRY 1	788 MWe	PWR	1/73

PLANT_NAME	NET_RATING	TYPE	COMMERCIAL:
SURRY 2	788 MWe	PWR	5/73
SUSQUEHANNA 1	1065 MWe	BWR	7/83
THREE MILE ISLAND 1	819 MWe	PWR	10/74
THREE MILE ISLAND 2	906 MWe	PWR	1/79
TROJAN	1130 MWe	PWR	6/76
TURKEY POINT 3	693 MWe	PWR	1/73
TURKEY POINT 4	693 MWe	PWR	10/73
VERMONT YANKEE	514 MWe	BWR	12/72
ZION 1	1040 MWe	PWR	1/74
ZION 2	1040 MWe	PWR	10/74

APPENDIX 2
F. R. G. PLANTS USED IN THIS STUDY

PLANT NAME	NET RATING	TYPE	COMMERCIAL:
BIBLIS-A	1146 MWe	PWR	3/75
BIBLIS-B	1240 MWe	PWR	1/77
BRUNSBÜTTEL	771 MWe	BWR	2/77
GRAFENRHEINFELD	1235 MWe	PWR	6/82
ISAR-1	907 MWe	BWR	3/79
NECKARWESTHEIM-1	795 MWe	PWR	12/76
OBRIGHEIM	340 MWe	PWR	3/69
PHILLIPSBURG-1	864 MWe	BWR	2/80
STADE	630 MWe	PWR	5/72
UNTERWESER	1230 MWe	PWR	10/79
WÜRGASSEN	640 MWe	BWR	11/75

APPENDIX 3
OPEC-2 CAUSE CODES

OPEC II CAUSE CODES

First Level / Second Level / Third Level

Note: 1. Code check valves "1" in the Third Level.

01 Undefined Failure

02 Fuel and Core 10

- 1 - Miscellaneous
  - 1. Miceallaneous fuel (such as PWR preconditioning)
- 6 - Core/fuel problems
  - 2. Burnable poison problems (e.g., BPRA vibration B&W -1978)
  - 3. Fuel failures/RCS activity (PWR)
  - 4. Foreign object
  - 5. BWR PCIOMR (Event No. must follow event this is associated with)
- 7 - Operational Restrictions
  - 4. Poison curtain changes (BWR)
  - 6. Control rod repatch (PWR esp. B&W)
- 8 - Mechanical Restrictions
  - 1. Increase core D/P (NH_4OH addition) (crud accumulation) 9
  - 2. Poison curtain vibrations (BWR-VY+PII-1973-1974)
  - 3. LPRM vibrations (BWR-4 1973-1976)
  - 4. Fuel failure - offgas limits (BWR)
  - 5. Fuel densifications
  - 6. Control Rod Guide Tube nut (B&W 1981) 10
  - 7. Control Rod Guide Tube (CE 1977) 10
- 9 - Safety Restrictions
  - 2. BWR control rod changes (includes fuel soak)
  - 3. ECCS peaking factor (PWR)
  - 4. EOL scram reactivity/rod worth restrictions (includes shutdown margin)
  - 5. Core tilt/Xenon restriction (out of flux band)
  - 6. BWR thermal limits (includes "rod limited")
  - 7. Thermal power restriction
  - 8. Reactivity coefficient (e.g., mod.temp. coeff.)

03 Reactor Coolant System

- 2 - Pumps
  - 2. Reactor coolant/recirc pumps and motors (except motor oil cooler-0363)
- 3 - Piping/Tanks
  - 2. Auxiliary piping (1-inch or less, vents, drains)
  - 3. Main process piping (include HPI nozzle/safe end crack - B&W 1982 and HPI thermal sleeve crack)
  - 4. Flanges, manways, fittings
  - 5. Supports, snubbers

OPEC II CAUSE CODES, (Cont.)

First Level / Second Level / Third Level

- 6. Strainers, filters
- 7. Core spray piping - IGSCC (no safe-end problems-0339/other nozzle problems-0373)
- 8. Recirc valve bypass piping (BWR beginning in 1974 - IGSCC)
- 9. Safe-end problem (BWR - IGSCC)

- 4 - Valves
  - 2. Auxiliary piping valves
  - 3. On-off valves (stop, stop-check, isolation)
  - 4. Control valves
  - 5. Relief valves (excluding primary system relief valves)
  - 6. Main steam isolation valve (BWR) (for PWR MSIV-0846)
  - 7. Primary system relief valves (PWR pressurizer; BWR main steam) RV, SRV, ARV, PORV, OPS
  - 8. Pressurizer spray valves
  - 9. Recirc. flow control
- 5 - Instruments & Controls
  - 2. Flow
  - 3. Temperature (BWR-main steam high temperature)
  - 4. Pressure & Overpressure protection system (OPS) 6
  - 5. Level
  - 6. Nuclear (ex-core detectors) (in-core/ex-core calibrations)
  - 7. In-core detectors (TIP, LPRM, APRM, APDM, all BWR nuclear monitors) (no RBM, LPRM vibration) 8
  - 8. Loose part monitor/nuclear noise systems/RDS (B&W)
  - 9. Integrated control system (ICS) (B&W)
- 6 - Heat Exchangers/Fans
  - 2. Control rod drive cooling fan and heat exchanger
  - 3. Reactor coolant pump motor-oil cooler
- 7 - Reactor Vessel & Internals
  - 1. Other (including CSS Sparger) 10
  - 2. Flanges/Seal Ring
  - 3. Vessel nozzles (excluding safe-end problem - 0339) thermal stress (see also 0379 - BWR CR nozzles)
  - 4. Internals (e.g., holddown spring - CE; thermal shield-W)
  - 5. Feedwater spargers (BWRs - 1972-1976)
  - 6. Jet pumps (BWR-3s 1973-1975)
  - 7. Specimen holders SSHT (B&W 1976)
  - 8. Control rod guide tube pin (Westinghouse 1982) 10
  - 9. Control rod nozzles thermal stress (BWR - see also 0373) 10
- 8 - Control Rod Assemblies
  - 2. Motor drives/magnetic jack drives (PWR)
  - 3. Hydraulic drive (BWR)
  - 4. Scram mechanism - BWR accumulator & pilot valve - ATWS I&C (see also 1295)
  - 6. Control rods (BWR-1973-1974 inverted CR; cracking)
  - 7. MG sets/logic/relays/power supply/sequence controller 6
  - 8. Control rod position indication

OPEC II CAUSE CODES, (Cont.)

First Level	Second Level	Third Level	
	9 - Water Quality/Chemistry		
		2. RCS chemistry	
		3. RCS chemical clean	
		4. RCS boron concentration	
04 Steam Generators			
	3 - Piping	(SG nozzle problem 1979 - see 9739)	9
		2. Auxiliary piping (1-inch or less, vents, drains)	
		4. Flanges manways, fittings, handholes	2
		5. Snubbers/Supports	
	4 - Valves (except blowdown system valves)		
		2. Auxiliary piping valves	7
		3. Off-on valves	
		4. Control valves	
		5. Relief valves	
	5 - Instruments and Controls		
		4. Pressure	
		5. Level	
		8. Loose Parts Monitor	9
	6 - Blowdown System		
		2. Supports	
		3. Piping	
		4. Valves	
		5. Instruments & controls for blowdown system	
	7 - Steam Generator Tubes		
		2. Caustic attack	
		3. Thinning	
		4. Denting (eddy current testing & tube support plate deterioration)	4
	8 - Other Components		
		2. Spargers	10
		3. Clad separation	
		4. Moisture separators/moisture carryover	
		6. Internal tube supports (excluding denting problem)	
		7. Clad problems (other than separation)	
	9 - Chemistry Modifications		
		2. Chemistry changeover (AVT-phosphate changeover)	
		3. Sludge lancing	
		4. Circulation modifications	
		5. Chemical cleaning	
		6. Boric acid soak	10
05 Chemical & Volume Control System/RX Water Cleanup System			
	2 - Pumps		2
		2. Charging (RSW Makeup Water Pumps)	
		3. Boric Acid Transfer Pump	
	3 - Piping/Tanks		
		2. Auxiliary piping	
		3. Main process piping	
		4. Flanges, fittings	
		5. Supports, snubbers	

OPEC II CAUSE CODES, (Cont.)

First Level	Second Level	Third Level	
		6. Strainers, filters	
		7. Tank (HUT, RAT or BAST, VCT)	2
		8. DWST, Primary Water Makeup Tank	7
	4 - Valves		
		2. Auxiliary piping valves	
		3. On-off valves	
		4. Control valves	
		5. Relief valves	
	5 - Instrumentation & Controls		
		2. Flow	
		3. Temperature (including heat tracing circuits)	3
		4. Pressure	
		5. Level	
		6. Chemistry	
		7. Boration	
	6 - Heat Exchangers		
		2. Regenerative	
		3. Nonregenerative	
		4. Excess letdown	
		5. RCP seal return	
	7 - Chemical Processing Equipment		
		2. Evaporator/concentrator	
		3. Gas stripper	
		4. Demineralizers (BWR - makeup system)(see 077)	
	9 - Chemistry		
		2. Boron concentration	
		3. Boron stratification in tank	
06 Condenser			
	3 - Pipes		10
	4 - Valves		10
	5 - Condenser Vacuum I+C		9
	7 - Tubes		
	8 - Loss of Vacuum and Back Pressure Limits (see 1679)		
	9 - Components		
		2. Shell/casing	
		3. Air ejector	
		4. Waterbox (e.g., fouling see also 1679)	
		5. Baffles	
		6. Staking (Palisades, PB-2)	
07 Condensate/Feedwater/Auxiliary Feedwater/Makeup Water Systems			•
	2 - Pumps & Pump Drives (elaborate in comments)		
		2. Feedwater (FW)	
		3. Condensate/booster pumps	
		4. Heater drain	
		5. Chemical/demineralizer	
		6. Other	
		7. Auxiliary feedwater	
	3 - Piping/Tanks		

OPEC II CAUSE CODES, (Cont.)

First Level	Second Level	Third Level		
		2.	Auxiliary piping	
		3.	Main process piping	
		4.	Flanges, fittings	
		5.	Supports, snubbers	
		6.	Strainers, filters	
		7.	Extraction steam system/heater drains/HDT	10
		8.	Demineralizer system	
		9.	FW&SG nozzles & safety portion of FW piping (I&E Bulletin 79-13)	3
	4 - Valves			
		2.	Auxiliary piping valves	
		3.	On-off	
		4.	Control	
		5.	Relief	
		6.	FW reg. valves	
		7.	Extraction steam	
		8.	Demineralizer system	
		9.	FW isolation valves	2
	5 - Instrument and Controls			
		2.	Flow	
		3.	Temperature	
		4.	Pressure	
		5.	Level (including FW heater level controls)	
		6.	Chemistry (e.g., chloride monitor)	
		7.	FW flow control (except B&W ICS-0339) (S/G high or low trips)	
	6 - Heat Exchangers			
		2.	Feedwater heaters	
		3.	Other (include FWP gland seal condenser)	
	7 - Demineralizers			
		2.	Capacity limitations	
	9 - FW Chemistry			
08	Main Steam System			
	3 - Piping/Tanks			
		2.	Auxiliary piping	
		3.	Main Process piping	
		4.	Flanges, fittings	
		5.	Supports, snubbers	
		6.	Strainers, filters	
	4 - Valves			
		2.	Auxiliary piping valves	
		3.	On-off	
		4.	Control	
		5.	Relief (MSRV & steam dump valve for PWRs) (MSRV for BWR-0347)	
		6.	Main steam isolation valves (PWR) and reverse checks (MSIV for BWRs-0356)	
		7.	Main steam bypass (to condenser) + BWR dump to condenser hotwell	3

OPEC II CAUSE CODES, (Cont.)

First Level	Second Level	Third Level		
	5 - Instruments & Controls			
		2.	Flow	
		3.	Temperature	
		4.	Pressure	
		5.	Level	
		6.	Moisture separator reheater	
		7.	Steam bypass/steam dump	6
	7 - Moisture Separator/Reheater (MSR)			
		2.	Drains/carryover/water level control problems	
		3.	Tubes/tube support problems (including tube leaks)	
		4.	MSR relief valve problems	
		5.	Other MSR valve problems	
		6.	MSR steam supply valve	
09	Turbine			
	3 - Piping			
		2.	Auxiliary piping	
		3.	Crossover piping	
	4 - Valves			
		2.	Auxiliary piping valves	
		3.	Intercept/stop	
		4.	Control/throttle/governor valves	
		5.	Combined intercept valves (CIV)	
	5 - Instruments & Controls			
		2.	Instruments	
		3.	EHC/supervisory system/ (GE-permanent magnet)	
		4.	Pressure Regulator (BWR-IPR EPR + MPR)	
		5.	Turbine Protective Devices (overspeed test)	9
	7 - Components			6
		2.	Shaft/blades	1
		3.	Bearings	
		4.	Gland seals	
		5.	Turning gear	
		6.	Casing	
		7.	Turbine Balancing	1
	8 - Oil System (do not include bearing or EHC problems)			
10	Generator			
	5 - Instruments & Controls			
		2.	Instruments	
		3.	Logic/controls (including under frequency relay)	
		4.	Core monitor (ionization detector)	
		5.	Voltage regulator	
	7 - Auxiliary Systems			
		2.	H_2 Cooling	
		3.	H_2O Cooling	
		4.	Oil (for bearings, control, seals)	
		5.	Bus duct/leads/bus duct cooling	10
	8 - Components			
		2.	Exciter (permanent magnet except in GE-EHC)	

OPEC II CAUSE CODES, (Cont.)

First Level	Second Level	Third Level	
		3. Rotor	
		4. Stator	
		5. Shaft	
		6. Bearings	
		7. Bushing	
		8. Turbine-generator-exciter shaft coupling	
11 Electrical Systems			
	7 - Transformers		
		2. Main	
		3. Other (startup, station auxiliary)	
	8 - Switchgear/buses (except instrument & safeguards buses)		
	9 - Safety-related equipment		
		2. Uninterruptable power supply (DC power system; 125 VAC instrument buses, inverters, MG sets, relays)	
		3. Emergency diesels (including output breakers)	
		4. Essential/vital safeguards buses (safeguards other electrical)	
		5. Electrical connectors	
		6. Gas turbines	
12 Reactor Trip System (Only failures of RTS or spurious trips not caused by actual trip parameters or detector/transmitter failures)			2
	6 - RCS Input Channels and Other Channels		2
		2. RCP breaker	
		3. Mode switch	6
	7 - Reactor coolant system (RCS) input channels (continued)		
		2. Nuclear instrumentation	
		3. RCS water level	
		4. Reactor pressure	
		5. RCS coolant flow	
		6. RCS temperature	
		7. Pressurizer pressure	
		8. Delta temp./low DNBR trip (CE - Thermal Margin + Low P)	
		9. Under voltage/under frequency trip	
	8 - Secondary system inputs		
		2. Turbine/generator inputs	
		3. Feedwater/steam flow mismatch and low steam generator level (steam & feed rupture control system)	4
		4. Main steam isolation valve closure	
		5. Main steam activity or temperature	
		6. Main steam pressure	
		7. Feedwater flow	
		8. Steam generator level	
		9. Main steam flow (high or low)	
	9 - Other channel/components		
		2. Containment drywell	
		3. Safety injection signal	
		4. RCS leak detection	
		5. Containment high pressure	

OPEC II CAUSE CODES, (Cont.)

First Level	Second Level	Third Level	
		6. Logic/relays/permissives	
		7. Manual	
		8. Reactor trip breakers (see also 0384)	
		9. Rod drop signal	
13 Auxiliary Systems			
	7 - Off-gas systems (AOG)/Ventilation Systems		10
		1. Offgas (excluding recombiners)	10
		2. Recombiner (BWR)	4
		3. Plant Vent/Filter System (PWR see also 2283), (SBGTS (BWR), (SLCRC-BV)	1
	8 - Other systems		
		2. Instrument/service air or nitrogen	
		3. Radioactive waste (RWCU) + RMS (area and process monitoring systems)	2
		4. Process computer/RWM/rod block/DDPS (at TP + SL)	4
		5. Auxiliary boiler	
		6. Fire Protection System (including fire barriers)	2
		7. Meteorological instruments	9
		8. Seismic instruments	9
14 Refueling/Maintenance			
	6 - Core physics tests		
	7 - Refueling		
	8 - Refueling equipment problems		
	9 - Maintenance		
15 Utility Grid (Noneconomic)			
	6 - Other off-site grid problems (grid maintenance)		
	7 - Loss of load/load rejection		
	9 - Loss of off-site power or off-site caused under-voltage condition or other electrical disturbance		
16 Circulating Water/Service Water System			
	2 - Pumps		
		2. Circulating water pump	
		3. Service water pump (BV - River Water System)	
		4. Cooling tower circulation pump	
		5. River Water Pumps (Farley) Aux RWS (BV)	6
	3 - Piping		
		2. Auxiliary piping	
		3. Main process piping	
		4. Flanges, manways, fittings	
		5. Supports, snubbers	
		6. Strainers, filters, fish & trash rake (see also 1672)	
	4 - Valves		
		2. Auxiliary piping	
		3. On-off	
		4. Control	
	5 - Instruments & controls		

OPEC II CAUSE CODES, (Cont.)

First Level | Second Level | Third Level

- 2. Flow
- 3. Temperature (for circ. water temperature limits - 1674)
- 4. Pressure
- 5. Level
- 6. Chemistry

6 - Heat exchangers
- 2. Cooling towers (no CT temp lim-1674/ultimate heat sink/safeguards CT-2362)
- 3. Tube/shell
- 4. Intake heaters

7 - Intakes/discharges (spray-pond problems)
- 2. Fouling/icing
- 3. Structural failure
- 4. Cooling water temperature/design insufficiency/ high temperature 10
- 5. Excessive fish kill
- 6. Discharge diffuser
- 7. EPA discharge limit 10

8 - Water treatment

17 Thermal Efficiency Losses

19 Core Cooling/Safety Injection System

2 - Pumps (containment spray pump - see 2223)
- 2. High pressure core injection
- 3. Low pressure core injection (except RHR pumps)
- 4. Residual heat removal (RHR) (no BWR-see LPCI)
- 5. Recirculation (I/ORSP - BV, S, NA) 7
- 6. RCIC (BWR)
- 7. Core Spray Pump (LPCS)

3 - Piping/tanks
- 2. Auxiliary piping
- 3. Main process piping
- 4. Flanges, fittings
- 5. Supports, snubbers
- 6. Strainers, filters
- 7. RWST & CST (RSW - BWST; BWR - DWST see 0338) 7
- 8. BIT 8
- 9. Accumulator (Core Flood Tanks, SIT, UHI)

4 - Valves
- 2. Auxiliary piping valves
- 3. On-off valves
- 4. Control valves
- 5. Relief valves
- 6. Check valves
- 7. SBLC explosive valve

OPEC II CAUSE CODES, (Cont.)

First Level | Second Level | Third Level

5 - Instruments & controls
- 2. Flow
- 3. Temperatures
- 4. Pressure
- 5. Level
- 6. Safety injection actuation (logic circuitry/actuators)
- 7. LPCI Loop Selection Logic

6 - Heat exchangers
- 2. RHR heat exchanger
- 3. Isolation Condenser (BWR)

9 - Chemistry

20 Initial Plant Startup/Operator Training
- 7 - Startup testing
- 8 - Power ascension/reduced power operation
- 9 - Operator Training/Emergency Plan Testing 7

21 Paired Unit Impact

22 Containment System (shield building)

2 - Pumps
- 2. Drywell pump
- 3. Containment building (quench) spray pump (no - I/ORSP-1923)(primarily on BWR-2 units) 7

3 - Piping
- 2. Auxiliary piping
- 3. Main process piping
- 4. Flanges, fittings
- 5. Supports, snubbers, high energy line problems, general snubber inspection
- 6. Strainers, filters

4 - Valves
- 1. Other valves (e.g., BWR vacuum breaker)
- 2. Auxiliary piping valves
- 3. On-off valves
- 4. Control valves
- 5. Relief valves (include PWR vacuum relief)
- 6. Containment building (CB) isolation valves (except MSIV & FW IV) 7
- 7. CB purge/exhaust valves (torus vent valve) 8
- 8. BWR MSIV leakage control system valves 7
- 9. Vacuum relief (BWR) 7

5 - Instruments & controls
- 2. Flow
- 3. Temperature
- 4. Pressure
- 5. Level

OPEC II CAUSE CODES, (Cont.)

First Level	Second Level	Third Level		
		6.	CB Isol actuation (VIAS)	
		7.	CB spray actuation (CDA)	5
		8.	Gas analyzer (BWR - Containment Inerting System) (PWR H-2 analyzer)(see also 2257)	10/4
	6 - Heat exchangers			
		1.	Drywell cooling	
		2.	Ice condensers	
		3.	Recirculation fan coolers (see also 2283)	
		4.	Casing Cooling System (NA - subatmospheric)	7
	7 - Containment structures			
		2.	Torus	
		3.	Penetrations	2
		4.	Containment leakage	
		5.	Drywell	
		6.	Liner (e.g., WCPS)	
		7.	Airlocks (including airlock seal leakage)	
		8.	Integrity (ILRT-App J Tests - 5th Tier-03) App J Test or unknown	5
	8 - Fans/air filters (Plant vent filters & SBGTS-1373)			
		2.	Recirculation air fans & dampers-PWRs (BWRs-drywell coolers)	
		3.	CB ventilation fans and ducts, and outside filters + H_2 purge (see also 1373 & 2263)	4
		4.	CB charcoal filters (inside CB)	
		5.	Vacuum pumps (sub-atmospheric containments)	
		6.	Penetration cooling fans	
		7.	H2 recombiner (PWRs)/blowers (see also 1372)	4
23	Component Cooling Water			
	2 - Pumps			
		2.	Component cooling water pump	
	3 - Piping			
		2.	Auxiliary piping	
		3.	Main process piping	
		4.	Flanges, fittings	
		5.	Supports, snubbers	
		6.	Strainers, filters	
	4 - Valves			
		2.	Auxiliary piping valves	
		3.	On-off valves	
		4.	Control valves	
		5.	Relief valves	
	5 - Instruments & controls			
		2.	Flow	
		3.	Temperature	
		4.	Pressure	
		5.	Level	
	6 - Heat exchangers			
		2.	Cooling towers, ultimate hat sink CT (no normal CT-1662)	
		3.	Tube/shell	

OPEC II CAUSE CODES, (Cont.)

First Level	Second Level	Third Level		
24	Structures + Intersystem Problems			2
	7 - Structures			
		2.	Control building (e.g., Trojan 1978 problem)	9
		3.	Auxiliary bldg. (e.g., Salem 1980)	9
		4.	Main steam tunnel (see also 1285 or 1373)	4
	5 - Electrical			
		2.	Cable routing	9
		3.	Cable splices and electrical connectors	10
		4.	Browns Ferry fire	10
		5.	San Onofre 1 cable fire	10
25	Economic			
	7 - Fuel economic			
		2.	Coast to refueling/fuel depletion	10
		3.	Fuel conservation	10
	8 - Grid economic			
		2.	Low-system demand/spinning reserve	10
		3.	Load following	10
99	Left Over			

FOURTH LEVEL: PROBLEM MODE/EXTENT

09 No failure, abnormality or malfunction
01 Unknown failure or problem undefined by below categories

Active Components (Pumps, Valves, Snubbers, Motors, etc.)
50 Total failure to operate, start or run (or just says failure)
51 Failure of automatic functioning but not manual
52 Failure of control (operates but erratic or incorrect control)
53 Mispositioning/misaligning/spurious closure (MSIV)/misoperation (e.g., erroneous start of pump). Note: component still working but did wrong thing 5
54 Performance degradation (operates out of spec)
55 Flow performance degradation/limits
56 Level degradation/limits
57 Time performance degradation/limits
58 Chemical degradation/limits
59 Radioactivity limits
60 Noncurtailing degradation (operable but nondisabling problem present)
61 Valve seat leakage
62 Structural/design/construction inadequacy 5

Passive System (Pressure Boundary, Pipes, Strainers)
65 Total failure (complete loss of flow)
66 Performance degradation
67 Flow performance degradation/limits (fouling/ice formation)
68 Level performance degradation/limits
69 Time performance degradation/limits
70 Chemical degradation/limits (water chemistry)
71 Radioactivity limits
72 Noncurtailing degradation (operable but nondisabling problem present)
73 Unspecified leak
74 Defective weld
75 Crack - no leak (stated)
76 Crack - leak
77 Tube leak
78 External leak from seal, gasket, packing, etc.
61 Valve seat leakage 3
79 Structural/design inadequacy/construction deficiency 5

Active Electrical Components (Generators, Relays, Breakers, Switch, Detectors, etc.)
80 Total failure to operate or perform (or just says failure)
81 Failure of automatic functioning but not manual
82 Failure of control (operates but erratic or incorrect control)
48 Mispositioning/Misoperation (see 4th Level, #53) 6
83 Performance degradation (operates out of spec)
84 Current degradation/limits
85 Time performance degradation/limits
86 Voltage degradation/limits (UV) (see also #93)
88 Setpoint drift 1
87 Noncurtailing degradation (operable with nondisabling problem present)
89 Spurious trip 3
49 Structural/design/construction inadequacy 5

FOURTH LEVEL: PROBLEM MODE/EXTENT, (Cont.)

Passive Electrical Systems (Buses, MCCs, Power Cables, Transformer)
90 Total failure (shorts, faults, grounds, arching, current interrupts from blown fuse resulting in complete loss of power)
91 Performance degradation
92 Current degradation/limits
93 Voltage degradation/limits (UV, voltage spike) (see also #86)
94 Noncurtailing degradation (e.g., cracks in insulation-corroded contacts) 5
95 Structural/design/construction inadequacy

Old Fourth Level Coding (may be encountered in some pre-1978 data)

Leakage/Crack
04 External leakage (packing/seals/gaskets)
15 Unspecified leakage
26 Tube leakage (HT exchanger/condenser)
31 External leakage - cracks
32 Defective weld
18 Valve seat leakage

Activity
05 High gaseous (external to plant)
06 High liquid
07 High process (in plant)
08 High area

Components
16 Valve operator
17 Valve component
18 Valve seat leakage
21 Relief valve lifting/failure to seat

19 Local control (pump to valve)

20 Pump components
22 Pump drive
23 Pump bearing
24 Pump M/G sets

33 Oil seal

Other
02 Water chemistry
03 Water temperature
27 Fouling - heat exchanger/intake/discharges, etc.
29 Structural failure
32 Defective weld

25 Electrical penetration welds

13 Offgas explosion

FIFTH LEVEL: EXTERNAL CAUSE OF EVENT

00	Not specifically specified as an external cause	
01	External cause-not covered below	

NRC Originated

02	Regulatory/Operational limit (Safety Limit of T.S.) (use only if outage results)	4
03	Regulatory requirement to inspect for possible deficiency	
04	Regulatory requirement to modify equipment due to malfunction or construction/design deficiencies	
05	Regulatory requirement to modify equipment due to more restrictive criteria	
29	NRC licensing proceedings and hearings	
19	Unavailability of safety-related equipment (use "02" if limit or restriction is known)	

Plant Originated External Causes

06	Testing (use for power red. or S.D. for test, not trips or failure during test)	
07	Testing error	
08	Maintenance error — these include procedural inadequacies	
09	Operator error	
59	Personnel involvement suspected to have precipitated problem/failure	2
26	Non-NRC precipitated derating (discretionary derating-no equipment failure)	
52	No failure/problem - purely preventive maintenance or inspection	
53	No mention of problem (possible preventive maintenance or NRC required mod - no degradation stated)	2

External Equipment Malfunction

11	Malfunction of local water supply	
12	Malfunction of local instrument air/service air supply/cooling air	
13	Malfunction of local oil supply	
14	Malfunction of local electrical supply	
17	Fuel supply	2
18	Local control/instrumentation (part of component package)	4
15	Other auxiliary system malfunction (e.g. cooling)	3

BWR Fuel Limits

30	CR seq. or pattern change (every 3 EFPW)	
31	CR Adj. Extension, etc.	
32	Periodic reduction for testing at CR adj. frequency (in BWR-TV test-1/month) (no mention of CR adjustment)	
33	Periodic reduction for load following at CR adj. frequency	
34	Periodic reduction for unknown cause at CR adj. frequency	
35	Periodic reduction for other reason at CR adjust frequency (no mention of CR)	6
36	Weekly reduction for testing	
37	Weekly reduction for other reason (suspected testing)	
40	MCPR	
41	MAPLHGR	
42	General thermal limit or comb. of 40 & 41	
43	Persistent undefined messing around between 80% & 100%	
44	Load drop (suspect fuel thermal limit)	

FIFTH LEVEL: EXTERNAL CAUSE OF EVENT (Cont.)

Cracking

71	Vibration induced crack	10
72	SCC induced crack	10
73	Thermal fatigue induced crack	10

Other

24	Plant internal or external environmental effects (lightning, icing)	
25	Fires	
61	Bearing malfunction	10
62	Pump or valve-drive mechanism	10

Combination

91	Combinations	62 (drive) & other 5th-level coding	10
92	"	07, 08, 09, 59 (error) & other 5th-level coding	10
93	"	02, 03, 04, 05 (NRC) & other 5th-level coding	10
94	"	3X, 4X, (BWR fuel) & other 5th-level coding	10
95	"	61 (Bearing) & other 5th-level coding	10
98	Multiple combination of 91, 92, 93, 94		10
99	Other level 5 combination		10

SIXTH LEVEL: METHOD OF SHUTDOWN

NOTES:

1. Use coding of method of shutdown reported in Gray Books (except for Classification 4 and 9).

2. Rather than using the Gray Book Classification 4, a consistent coding of the method of shutdown should be used for each outage if it extends into multiple monthly periods (unless the status changes - e.g., reduction to shutdown). 2

3. Classifications marked with an asterisk are different than those in the Gray Books.

4. The number of startups equals the number of outages of Classification 1, 2, and 3.

0 Not an outage or reduction (work which is done with unit operating and no derating - e.g., LERS which occur during operations but cause no shutdown) 2
1 Manual (includes manual shutdowns with a unit trip at low power
2 Manual scram
3 Automatic scram (including test performed with intent of tripping unit)
4 Turbine trip with no reactor trip* (reactor trip is assumed unless told differently) 2
5 Load reductions 9
6 Continuation of other outage (no interstitial electrical generation)*
7 Noncurtailing or concurrent work or inspection within an outage* (including all ROs during an outage)
8 Related transient (ie., second failure that follows from events of an earlier failure or abnormal functioning during the earlier failure)*
9 Unknown*

SEVENTH LEVEL: SHUTDOWN PARTICULARS

0 No extra information or no shutdown
1 SI signal generated (only one per SI generated--with most immediate cause)
2 There are related transients or significant subsequent events following this
3 event (must always precede event with 8 in 6th level)
4 2 + 3 7
5 The outage/incident results from a test/inspection/test failure (not an operational failure)
6 Scheduled test performed with intent of tripping the turbine or reactor (see #3, Level 6)
7 3 + 6 2
8 3 + 5
9 3 + 2

EIGHTH LEVEL: SIGNIFICANT TRANSIENT INITATOR

NOTES:

1. Significant transient initiating events or precursors should be denoted in this code classification.
2. Initiating events are to be coded (i.e., are significant) if they result in or would have resulted in a trip or runback of the unit (e.g., loss of feedwater flow should be coded only if the reactor trips or a feedwater pump trips) except for those transients where no trip is expected (no operations out of spec without a resulting trip should be coded here).
3. Such significant transients identified in WASH 1400 are identified by an asterisk(*).
4. Multiple significant transient initiators in a single event should be denoted by a new code classification.

00 No significant transient initiator
01 Unknown or uncertain significant transient initiator
99 Unclassified as yet
02 Reactor trip (no other initiating events noted) + spurious reactor trips 2
03 Generator trip 9

Turbine/Generator Transients
05 Turbine trip (overspeed trip)*
06 Turbine trip (other)*
07 High steam flow* (no RV or Dump V stuck open) (see 77, 31)
08 Low steam flow*
42 Loss of load with subsequent loss of off-site power
43 Excessive load increase with subsequent loss of off-site power
44 Loss of load
45 Excess load increase*
46 Turbine trip with failure of generator breaker to open (failure to relay auxiliary loads to off-site power)* 4
47 Other

Electrical Power Transients
40 Loss of power on an auxiliary bus (4.9 kv, 480 v, 120 v)
41 Loss of AC power from off-site network* (station blackout) (include partial blackout) 5
42 Loss of load with subsequent loss of off-site power
43 Excessive load increase with subsequent loss of off-site power
44 Loss of load*
45 Excess load increase*
46 Turbine trip with failure of generator breaker to open (failure to relay auxiliary loads to off-site power)* 4
47 Other

Feedwater, Condensate, Circulating Water and CVC System Transient
11 Loss of FW flow - FW pump or pump drive problem*
12 Loss of FW flow - malfunction of FW flow control* (including valve failure and low SG level)
13 Increase of FW flow - malfunction of FW flow control* (including valve failure and high SG level)
15 Loss of condensate pumps*

<u>EIGHTH LEVEL: SIGNIFICANT TRANSIENT INITATOR</u>, (Cont.)

<u>Feedwater, Condensate, Circulating Water and CVC System Transient</u>, (Cont.)
16 Loss of circulating water pumps* (causing unit trip a runback)
17 Loss of condenser vacuum*
18 Loss of FW heating* (60°F step change)
19 CVCS or other malfunction resulting in RCS boron dilution* (no trip necessary)
91 Radiation release from CVCS or other system in auxiliary building (no trip necessary)

<u>Reactor Coolant Pump/Recirculation Pump Transients</u>
51 Trip caused by startings of inactive RC or recirculation loop
52 Recirculation flow control failure - decreasing flow (BWR)*
53 Recirculation flow control failure - increasing flow (BWR)*
54 RCP trip or malfunction* (including shaft break and partial loss of flow)
55 RCP seizure*
74 RCP seal failure

<u>Reactor Coolant System Pressure and Temperature Transients or SI or RXT</u>
56 Inadvertent depressurization of primary system (no leaks) (no trip necessary)
57 Inadvertent overpressurization of primary system (no trip necessary)
58 Excessive cooldown or heatup rate (no trip necessary)
59 High or Low RCS Temp. (no trip necessary) or T&P fluctuations 6
60 Pressure and/or level fluctuations including bubble in RCS (no trip necessary) 9
63 Inadvertent SI or spurious SI signals (HPCI pump start - BWT)*
64 Accidental depressurization of the main steam system (e.g., stuck open RV) (see #34) (no trip necessary) 3
02 Reactor trip (no other initiating events noted) + spurious reactor trips 2

<u>Control Rod Transients</u>
61 Uncontrolled (for improper) rod (assembly or bank) withdrawal at power*
62 Uncontrolled rod withdrawal during startup*
63 Control rod assembly drop or misalignment* (no trip necessary)
76 Rod ejection or CRDM housing rupture

<u>Valve Malfunction Transients</u>
31 Malfunction of control resulting in inadvertent opening of a turbine steam bypass valve* (see 07,77) (or stuck open bypass valves) 7
32 Closure of MSIV*
33 Spurious opening of S/RV (RCS for PWR, MS for BWR)* (no trip necessary)
34 Stuck open S/RV (see 66)
35 Spurious opening of S/G PORV (PWR)* (no trip necessary)

<u>Reactor Coolant System/Steam Leak</u> (LOCA and Steam Break)
71 Large RCS leak (from equivalent 6-inch diameter hole or larger)*
72 Small RCS leak*
34 Stuck open S/RV
73 SG tube leak* (PWR) (see 9th L 031)
74 RCP seal failure
75 RCS system boundary valve failure* (leak past valve - see #72) (9L #062) 3
76 CRD mechanism housing rupture or CP ejection
77 Steam line break* (all sizes) (see 07,31)
78 FW pipe break* (all sizes)

<u>EIGHTH LEVEL: SIGNIFICANT TRANSIENT INITATOR</u>, (Cont.)

66 Accidental depressurization of the main steam system (e.g., stuck open RV) (no trip necessary)
97 High steam flow* (no RV or Dump V stuck open) (see 77,31)
98 Low steam flow*

<u>Other Transient Initiators</u>
85 Fuel misloading (no trip necessary)
86 Refueling accident (e.g., dropped fuel assembly or dropped RV component) (no trip necessary)
87 Shipping fuel pool accident (e.g., cracked pool liner) (no trip necessary)
88 Shipping cask storage tank release (no trip necessary)
89 Waste gas storage tank release (no trip necessary)
90 Liquid waste storage tank release (no trip necessary)
91 Radiation release from CVCS or other system in auxiliary building or outside CB (except 89 and 90) (no trip necessary)
92 Environmental (tornadoes, flooding, fire, earthquake) (no trip necessary)
93 Control room uninhabitability (9th Level - 190) (no trip necessary)
94 Loss of Service Water System (9th Level - 981)
95 Loss of Instrument Air System (9th Level - 191)
96 Loss of RHR flow (no trip ncessary)

NINTH LEVEL: SIGNIFICANT SAFETY SYSTEM PERFORMANCE

NOTES:

1. A failure unless otherwise specified is defined as follows:

   a. System failure of a system to meet minimum design requirements (e.g., minimum flow through proper paths).

   b. Component failure - failure of the component to perform as intended or meet specifications. Examples:

      (i) Valve failure - failure to operate, or in wrong position, or leakage past valve
      (ii) Tank failure/incapacitation - leak, or level, or concentration out of spec (inconsistent with specifications)
      (iii) Pump failure - failure to start, keep running, or deliver design flow

2. If there is a component failure that results in a system failure as well, the outage should be coded as a system failure.

3. Multiple failures or failures of safety systems not covered by the following coding classifications should be given a new code number.

4. Most LERs that are coded should have an entry in this level.

000 No unconservative failure in safety system
001 Unknown

999 Unclassified
008 No failure - procedural inadequacy
009 No failure - QA inadequacy 8

Unanticipated or Common Mode Safety Event
011 Nonconservative errors in SAR accident analyses or Tech Spec bases
012 Common mode incapacitating of safeguards equipment (describe in comments)
013 Disagreement with predicted value of reactivity balance (LER Prompt Report #4)
014 Structural inadequacy potentially affecting safety 2
015 Piping inadequacy potentially affecting safety
032 Snubber out of compliance
016 Electrical cables - potential inadequacy affecting safety 2
017 Other potential inadequacy affecting safety
018 TMI modifications 7

EP = Electric Power
021 System failure (insufficient AC or DC power to safeguard buses to operate minimum ESS features)
022 Diesel generator failure (failure to start on demand, while running, to load on bus, or of a support system that would incapacitate the DG) 2
023 Gas turbine failure/hydro unit failure (Oconee)
024 DC power supply failure (battery, inverter, etc.)
027 Safeguards electrical failure (other than bus) 2
028 Offsite power source problem 9
025 Safeguards bus failure
026 Instrument bus failure
028 Other offsite problems (no Oconee hydro unit) 9

NINTH LEVEL: SIGNIFICANT SAFETY SYSTEM PERFORMANCE, (Cont)

RTS = Reactor Trip System (or ROS)
031 System failure (failure of more than 2 full-length CR to insert)
032 Uninserted CR following trip
033 RTS I&C or logic unconservative malfunction (no spurious trip - or tripped channel) (refer to LER Prompt #1, 30 Day Report #1) (including detector)
035 CRD or CRD system problem (no dropped CR) 10/9

SI or ESF Actuation
036 System failure
037 Logic or I&C unconservative malfunction (including detectors) 3

Containment and Secondary Containment Problems
040 Other CB failure - not listed below (e.g., low torus DP) 2
041 Large leak (Xcfm) in airlocks, penetrations, etc. + system failures in CIAS 5
019 Other containment subsystem system failure 5
042 Small leak (Xcfm) in airlocks, penetrations, etc.
043 Containment Isolation System I&C, logic, actuating circuitry failure
044 Containment isolation valve failure and leaks (including ventilation valves)
045 Containment vent/purge - Standby Gas Treatment System (SBGTS) failure
046 Hydrogen recombiner failure
047 Weld Channel Pressurization System component failure (PWR)
048 Containment recirculation fan/cooler/filter system component failure (PWR) Drywell Cooler (BWR)
049 Post-accident pressure relief system component (PWR) + Vacuum breaker (BWR)
032 Snubber or piping support out of compliance of failure
033 MSIV failure (including leaks) 8
037 BWR MSIV leakage control system (valves & controls) 7
034 FW isolation valve failure
035 CB vacuum pump failure 2
264 Containment Depressurization Actuation I&C failure (PWR)
307 ADS actuation problems (BWR)
036 CB Inerting System component failure (include gas/H2 analyzer) 2

NOTE: For CB Spray see 330s for BWRs and 260s for PWRs 3

Safety System Boundary Abnormal Degradation (LER Reportable Item Prompt #3, Delayed #4)
061 In fuel cladding
062 In reactor coolant pressure boundary (including RCS - not CB - isolation valve leak)
(see above) In primary containment
063 In other container of radioactivity
064 Loose part in RCS 10
065 Loose part in Steam Generator 10

Safety System Coolant System (PWR = CHRS or Component Cooling Water; BWR = HPSW)
071 System failure
072 CHRS/HPSW pump failure
073 Heat exchanger failure
074 System valve failure
075 System I&C failure
076 Other failure

NINTH LEVEL: SIGNIFICANT SAFETY SYSTEM PERFORMANCE, (Cont)

Emergency Service Water System (Level 8-94) (NSRW) (RV - River water pumps)

081 System failure
082 Emergency service water pump failure
083 System valve failure
084 System I&C failure
085 Other failure

Ultimate (Safeguards) Heat Sink

091 System failure
092 System valve failure
093 System I&C failure
094 Other failure

Service Systems

101 Instrument Air System component failure (Level 8-93)
102 Fire Protection System component failure
103 Failed fuel detector failure
104 CCR habitability systems component failure (Level 8-93)
105 Radiation Monitoring System component failure
106 Filter Exhaust System outside of CB - HVAC (BWR area coolers)
045 SBGTS (BWRs)

Other

120 Turbine stop valve (TSV) or CV does not close or slow response

ESS (EMERGENCY SAFEGUARDS SYSTEMS)

PWR

ECI (Emergency Coolant Injection)

211 System failure (failure of accumulator, LPIS, HPIS, etc.)
212 Refueling Water Storage Tank (RWST) failure/incapacitation
213 Heat tracing system component failure
214 RCS leak detection system

LPSI Failures

221 Accumulator failure/incapacitation
220 RHR pump failure
222 LP pump failure
223 System valve failure
224 I&C failure
225 Other failure

HPIS Failures

230 Charging pump failure
231 HP pump failure
232 Boron Injection Tank (BIT) failure/incapacitation
233 Boric Acid Tank (BAT) failure/incapacitation
234 System valve failure
235 I&C failure
236 Other component failure of HPIS
237 Other component failure of CVCS or Boron Addition System
238 Upper head injection system valves
239 Upper head injection accumulator and other

LPRS (Low Pressure Recirculation System)

241 System failure
242 System valve failure
243 I&C failure
244 Other failure

ECR (HPRS) (Emergency Coolant Recirculation for Small Break LOCA)

251 System failure
252 System valve failure
253 I&C failure
254 Other failure

CSIS (Containment Spray Injection System) (CSS)

261 System failure
262 Containment spray pump failure
263 System valve failure
264 System I&C failure (include CDA)
265 Other failure

CSRS (Containment Recirculation System) (I/ORSP)

271 System failure
272 Containment recirculation pump failure
273 System valve failure
274 System I&C failure
275 Other failure

SHA (Sodium Hydroxide/Hydrazine Addition)

281 System failure
282 NAOH (or hydrazine) tank failure/incapacitation
283 System valve failure
284 System I&C failure
285 Other failure

AFWS (Auxiliary Feedwater System)

291 System failure
290 Unspecified AFW pump failure
292 Small AFW pump failure
293 Large AFW pump failure
294 Condensate Storage Tank (CST) failure
295 System valve failure
296 System I&C failure
297 Other failure

Safety and Relief Valves (S/R)

301 RCS pressurizer code safety valve failure to open
302 RCS pressurizer code safety valve failure to reseat
303 RCS pressurizer S/R valve failure to open
304 RCS pressurizer code S/R valve failure to reseat
305 RCS pressurizer Overpressure Protective System failure
311 RCS pressurizer safety other problem
312 RCS pressurizer relief valve other problem
306 MS safety valve failure to open
307 MS safety valve failure to reseat
308 MS relief valve failure to open
309 MS relief valve failure to reseat
310 MS SRV other problem

RHR System

321 RHR System failure
229 RHR pump failure
322 RHR valve failure
323 RHR I&C failure
325 Other RHR failure

ESS (EMERGENCY SAFEGUARDS SYSTEMS), Cont.

BWR

ECI (Emergency Coolant Injection)
611 System failure not listed below 9
612 Condensate Storage Tank (CST) failure/incapacitation
613 MUD-DSWT
614 Leak detection system component problem 7

LPCIS + RHR Failures (LPCI/RHR/VCCS)
621 Pump failure
622 System valve failure
623 System I&C failure
624 Other failure
625 RHR heat exchanger problem 7

CSIS Failures (CS)
631 (LP) core spray pump failure
632 System valve failure
633 System I&C failure
634 Other failure

HPCIS Failures
641 HP core injection pump failure
642 System valve failure
643 System I&C failure
644 Other failure

Standby Liquid Poison Injection System
651 SLPI system failure
652 SLPI tank failure
653 SLPI pump failure
654 Explosive plug valve failure
655 System I&C failure
656 Other failure

RCIC Failures + Isolation Condenser System Failures 3
661 RCIC pump failure
662 Isolation condenser failure
663 System valve failure
664 System I&C failure
665 Other failure

VS (Vapor Suppression) System
681 System failure
682 Torus support cracks
683 Other torus problem
684 Torus DP problem
685 Vacuum breaker problem (drywell to torus) 1
686 Other VS problem
687 Vacuum relief problem (CB to RaB) include permissive problems 7

FW (Feedwater System
691 System failure
692 Feedwater pump failure
693 Condensate pump failure
694 System valve failure
695 System I&C failure
696 Other failure

S/R (Safety and Relief Valves) + ADS
501 MS safety valve failure to open
502 MS safety valve failure to reseat
503 MS S/R valve failure to open - manual operation
504 MS S/R valve failure to open - automatic operation
505 MS S/R valve failure to reseat
506 MS S/R valve other performance failure 8
507 ADS actuation problems 7

Other
520 Component failure in dump to condenser Hotwell System 3

Containment Spray System (BWR-2s)
530 CSS System failure
531 CSS pump failure
532 CSS valve failure
533 CSS heat exchanger failure 3
534 CSS I&C failure
535 Other CSS component failure

Appendix 4

Glossary of Acronyms

AE	Architect/Engineer
AEC	Atomic Energy Commission
ASME	American Society of Mechanical Engineers
BMI	Federal Ministry of the Interior
BWR	Boiling Water Reactor
CEO	Chief Executive Officer
CFR	Code of Federal Regulations
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
GE	General Electric Company
GRS	Reactor Safety Company
IEB	Inspection and Enforcement Bulletin
IGSCC	Intergranular Stress Corrosion Cracking
INPO	Institute of Nuclear Power Operations
ISP	Integrated Schedule Program
KTA	Nuclear Safety Standard Commission
KWU	Kraftwerk Union
LWR	Light Water Reactor
MWe	Megawatts Electric
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PRA	Probabilistic Risk Assessment
PUC	Public Utility Commission
PWR	Pressurized Water Reactor
RSK	Reactor Safety Commission
SALP	Systematic Assessment of Licensing Performance
TMI	Three Mile Island
TUV	Technical Inspection Agency
VGB	Association of Large Power Producers