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An International Comparison of the Impact
of Safety Regulation on LWR Performance

by

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MIT-EL 87-006 June 1987

# AN INTERNATIONAL COMPARISON OF THE IMPACT OF SAFETY REGULATION ON LWR PERFORMANCE

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B.S. Mech. Eng., Cornell University
(1985)

SUBMITTED IN PARTIAL FULFILLMENT
OF THE REQUIREMENTS FOR THE
DEGREE OF

MASTER OF SCIENCE

IN TECHNOLOGY AND POLICY

at the

MASSACHUSETTS INSTITUTE OF TECHNOLOGY

June 1987



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ABSTRACT

During the decade 1975-1984, the US nuclear power industry achieved a lower level of reactor performance than that realized in many other Western nations. Previous work suggested that international differences in safety regulation account for much of the discrepancy. US annual regulatory losses averaged over 10% during the ten-year study period. The present investigation compares nuclear safety regulation in France, Sweden, and Switzerland with that in the United States 1) to determine whether greater regulatory stringency was indeed responsible for poorer US plant performance, and 2) to examine key international differences in the the division and coordination of responsibility between safety regulators and nuclear utilities for recognizing and solving technical problems.

Analysis of the US data revealed that, on average, over 90% of US regulatory outages were attributed to one of the following: technical specification limiting conditions of operation or NRC-required inspections or NRC-required modifications. It was found that the European nations experienced the same variety of technical problems seen in the United States. Furthermore, the scope and stringency of European and US safety regulation are comparable. It was found that inconsistencies in outage reporting practices account for much of the discrepancy in regulatory loss between the United States and the other nations. Therefore, it is concluded that safety regulation is not the primary cause of differences in reactor performance observed between the United States and other nations.

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Title: Professor of Nuclear Engineering

ACKNOWLEDGEMENTS

Many individuals and organizations deserve acknowledgement and sincere thanks for the support and assistance generously provided throughout the various phases of this project.

Financially, this research was supported by the US Department of Energy, the Westinghouse Electric Corporation, the Center for Energy Policy Research at MIT, and the Technische Universität Berlin.

The interviews with nuclear industry and regulatory officials were central to the investigation; many people gave their time in scheduling meetings, speaking with us, and reviewing drafts of this paper. We are indebted to these individuals for the information and comment provided. Special thanks to Mr. Pierre Lienart of EdF and Mr. Bengt-Göran Olofsson of KSU for their many contributions, including coordinating our visits to their respective countries.

Much of this project was completed while studying at the Technische Universität Berlin (TUB) in early 1987. Financial support for my visit was provided by a grant from the Stiftung Luftbrückendank (Airlift Gratitude Foundation). Special thanks to Frau Roswitha Paul-Walz in the Foreign Students' Office at TUB for her kind and cheerful assistance with the logistics of my stay.

The Division for Energy Engineering and Resource Economics at TUB was my academic home while in Berlin. I am grateful to Professor Dietmar Winje, for his personal assistance and contributions to the research. Thanks also to my colleague Dipl.-Ing. Ulrich Lorenz for his daily assistance and many interesting discussions. The Institute's technical assistants, secretarial staff, and student employees provided invaluable and friendly aid with telephone calls, mailings, and German computer software. All these people made my visit to Berlin professionally successful and personally most enjoyable.

At MIT, many thanks to my advisor, Professor Kent Hansen for his valuable comments and guidance throughout this writing and research. Professor Richard Lester also contributed importantly to early discussions for the project.

Finally, I would like to thank Jeff Jacob and Rachel Sheffet in Boston and my parents, Carolyn and Jerry Anderson in Washington, DC, for their personal assistance in running errands, making inquiries, and waiting in line on my behalf while I was away. Their

efforts and their regular letters were greatly appreciated.

S. C. A.
May 1987
Berlin (West)

To my parents, for their
love, patience, and support
throughout my life

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# 1.0 Introduction

## 1.1 Background

Most light water reactors (LWRs) used around the world for electric power production share substantially the same design and technology. As a result, the same types of technical problems with these power plants are experienced internationally. In light of this, it is curious that the performance of these nuclear plants varies widely from nation to nation. In the United States, for instance, performance is significantly inferior to that achieved by many other Western nations. This performance discrepancy likely arises from varying human influences rather than from technical factors. For example, nuclear plant management policies and regulatory organizations assume a widely different character from nation to nation. Through an international comparison of such human structures, insight may be gained into the underlying causes of poor plant performance in the United States.

Earlier work comparing the Federal Republic of Germany with the United States identified regulation as a chief contributor to the performance discrepancy between these two nations.[1] Another previous investigation[2] revealed that the United States reports significantly higher values of regulatory loss* than are observed in many other nations. Table 1.1 on page 3 gives data for countries previously investigated for the ten-year period 1975-1984.

From the figures presented in Table 1.1, it is tempting to infer that the NRC, as the US safety regulator, enforces a more stringent set of requirements than do the regulators in the other nations. Some US utilities believe this to be true, blaming burdensome regulation as a significant cause of poor nuclear plant performance. Before reaching this conclusion, however, one must verify that all nations studied have used the same definition of regulatory loss in their reported statistics.

This investigation attempts to account for the international differences observed in regulatory losses, such as those in Table 1.1. The inquiry centers around the question of consistency among nations in defining what does and what does not constitute a regulatory loss. Without a uniform definition for these losses, one is left comparing the

*For the uninitiated, an "annual loss" in the vernacular of performance statistics may be crudely defined as the total fraction of time that plants are shut down each year. Regulatory losses (i.e., losses due to the requirements of a regulatory body) are one component of this annual loss. More detailed information on performance statistics is given in Section 1.3, "Measuring Nuclear Power Plant Performance."

TABLE 1.1: Average Annual Regulatory Loss, for Six Nations, 1975-1984 (in percent)

| Nation | PWRs | BWRs |
|---|---|---|
| Federal Republic of Germany | 0.9 | 11.3 |
| France | 0.0 | NA |
| Japan | 0.0 | 0.0 |
| Sweden | 4.4 | 0.7 |
| Switzerland | 0.0 | 0.0 |
| United States | 10.9 | 10.4 |

(PWR = pressurized water reactor;
BWR = boiling water reactor;
NA = not applicable -- France had no BWRs included in the study)

Source: Wilson, pp. 290-92.

incomparable; reasonable inferences may not be drawn from the data of Table 1.1. Once consistency is assured, the relative impact of safety regulation upon plant performance in these nations will be clear.

An earlier investigation applied consistent outage classification criteria to data from the Federal Republic of Germany and the United States. Upon reclassification, the rather surprising result was that German regulatory losses (for all plants, PWRs and BWRs) exceeded those in the United States, when compared on an equal footing.*[,3] The present investigation will analyze outage classification conventions in France, Sweden, and Switzerland and compare the practice of these nations to that of the United States. As indicated in the opening paragraph, the key difference among these nations is probably not reactor technology, but rather the responses of nuclear utilities and safety regulators to the complexities and problems of this technology.

Thus, two avenues of inquiry (somewhat intertwined) are proposed:

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*Citing Hulkower's results for regulatory loss from all US and German plants:
As stated: US 10.7% ; FRG 4.4%
After reclassification: US 8.7% ; FRG 10.3%.

- Examine the nature of outage classification in France, Sweden, and Switzerland vis à vis that in the United States.

- Examine the relationships among the relevant actors in each nation's nuclear safety regulatory system, i.e., utilities, regulators, central government, and public intervenors.

The organizational environment is significant in the study of outage classification. In particular, the division of responsibility between utilities and their safety regulators will influence the rationale for outage classification in each nation.

With an understanding of outage classification conventions in the three European nations, a proper comparison may be made with US practices. Through this process, the unique burden of safety regulation on US nuclear reactor performance will be revealed.

## 1.2 The Importance of Good Performance

Nuclear power plant performance is not just a matter of jargon and statistics. Indeed, there are substantial costs, monetary and otherwise, incurred because of poor performance.

The most tangible and immediate cost is the cost of replacement power. Aside from hydropower, nuclear plants are the least expensive baseload units for a utility to operate (note that the low expense partially explains the baseloading).[4] When nuclear facilities are not available to meet demand, power must be obtained from higher cost plants, usually oil-, coal-, or gas-fired. Another monetary cost, less immediate but just as real, is that new plants will have to be built sooner if the performance of existing plants is worse than expected. Conversely, as Wilson implies,[5] a substantial improvement in performance could actually forestall new plant construction. Faced with narrowing reserve margins as demand increases faster than expected, many US utilities are keenly interested in any alternatives to building new plants.

A potential cost of poor performance, primarily economic in nature, is that a great number of plant start-ups and shutdowns could hasten the deterioration of many plant systems. This is not to say that plants having such an operating history are unsafe, although some safety margins may well be narrowed by the cyclical stresses on the plant. The surest effect, rather, is that the service lives of affected components will be shortened; more frequent replacements will be required, thus raising maintenance costs. Furthermore, the entire plant's lifetime may be abbreviated if mothballing becomes a less expensive alternative than continued operation with extensive, ongoing component replacement.

A non-monetary cost of poor performance is the increase in radiation exposure for utility maintenance personnel. A significant number of outages involve inspection or maintenance tasks conducted within the containment building. Naturally, many radiological safety precautions are taken to minimize the dose received by the people performing the work. Nonetheless, fewer such outages will mean less radiation exposure for utility maintenance staff.

## 1.3 Measuring Nuclear Power Plant Performance

A variety of criteria are employed from nation to nation to gauge nuclear power plant performance around the world. These may be broadly divided into two classes: 1) load factor related, and 2) energy availability related. For this investigation, two such indices are important in representing the data of the four countries studied. The capacity factor (CF) is a common load-related criterion; the energy availability factor (EAF) is a criterion of the second type. These are defined below:

$$CF = \frac{NEG}{NER * PH}; \qquad (1.1)$$

$$EAF = \frac{\left(\frac{NEG}{NER} + EEDH\right)}{PH}; \qquad (1.2)$$

where

CF = Capacity factor
EAF = Energy availability factor
EEDH = Equivalent economic derating hours: the total equivalent full power hours lost for economic reasons, e.g., load following, fuel conservation, coastdown to refueling
NEG = Net electrical generation (MW)
NER = Net electrical rating (MW)
PH = Period hours: for annual CF or EAF, the number hours in a year

(Any internally consistent set of units is acceptable.)

In nations that baseload their nuclear reactors, the economic losses (EEDH in equation 1.2) are negligible or zero. In this case, the CF and the EAF values are very close. This is, in fact, true in most nations; France is a notable exception. Over two-thirds of the French electricity supply comes from nuclear sources. With this substantial nuclear fraction, some load following must be practiced with nuclear plants. This policy results in a 4.6 percentage point difference between CF and EAF in the performance statistics of France.[6]

For this investigation, the choice of performance indices for each nation is shown in Table 1.2 below.

TABLE 1.2: Performance Indices Used in the Study

| Country | Performance Index |
|---|---|
| France | Energy Availability |
| Sweden | Capacity |
| Switzerland | Capacity |
| United States | Capacity |

For countries where the discrepancy between capacity and energy availability is significant (only France in this study), EAF is the criterion of choice. As the name implies, this parameter expresses the energy that is obtainable from the plant, whether or not the electricity is actually required to meet demand. EAF thus most accurately assesses the capabilities of power plants, as it distinguishes true performance potential from the effects of external economic and demand factors. In countries where the two indices are substantially identical, the choice of statistics was made according to the availability and completeness of data.

Throughout the report, references are made to "regulatory losses." If applied to a particular country, this term refers to either capacity or availability loss, according to Table 1.2. When not applied to a specific country, regulatory loss refers to the (negative) regulatory impact on performance, regardless of the means of measurement.

## 1.4 Outline of the Investigation

Chapter 2 outlines the nature of the US data, including regulatory outages in particular. For purposes of international comparison, this chapter identifies the major causes of US regulatory loss and the plant systems affected by these losses.

The next three chapters, 3, 4, and 5, discuss the results of interviews in France, Sweden, and Switzerland, respectively, with regulatory officials and utility industry representatives. Outlined for each country are the nature and behavior of organizations relevant to nuclear safety, principally safety regulators and utilities. Outage classification is also analyzed. Finally, comparisons are drawn with the US situation.

Conclusions are presented in Chapter 6. In particular, US regulatory outages are reclassified according to European conventions. Recalculated values of US loss are then compared to the figures for other nations. A summary discussion is given of regulatory and organizational differences between the United States and the European nations. Some recommendations for further work conclude the report.

## 2.0 Regulatory Impact on US Plant Performance

### 2.1 US Data -- Source and Method of Analysis

The source of US data for this study was a portion of the Operating Plant Evaluation Code - 2 (OPEC-2) database. OPEC-2 is maintained by the S. M. Stoller Corporation for the Institute of Nuclear Power Operations (INPO). The database incorporates all commercial LWRs in the United States larger than 400 MWe. All events at these plants which cause outages or which are otherwise significant are included in OPEC-2.* The data used for this study were all those events (a total of 37,492) occurring in the decade 1975-1984.

Central to the OPEC-2 database is the descriptive numerical coding for each event, hereafter referred to as the cause code. The cause code is a fifteen-digit cipher which describes each event with respect to the plant hardware affected, external influences on the event (both physical and regulatory), particulars of safety system operation, and miscellaneous other considerations. A cause code list is supplied in Appendix 8.1 showing the various coding levels which together fully describe each event. For this

*In addition to outages, significant events include: major repair or maintenance, safety system actuation or failure, and any event contributing to the critical path of an outage.

investigation, the database management software dBASE III by Ashton-Tate was used.

## 2.2 Regulatory Outage Classification

As the database manager, the Stoller Corporation uses data from several organizations, including utilities, equipment vendors, and the NRC in generating OPEC-2. Among these sources, a variety of outage classification schemes is in use. To ensure consistency in the database, Stoller applies uniform classification criteria in distingushing regulatory outages from purely technical problems. Thus, while the utilities have some influence on OPEC-2 outage classification, Stoller has the last word.

In the Cause Code List on page 110 of Section 8.1, Stoller lists the outage causes that it considers to be regulatory under the heading, "NRC Originated." Previous investigators had added some categories* also thought to be regulatory to this grouping;[7] their additions are preserved here. Out of the total of 37,492 US events from 1975-1984, this investigation selected 5,105 as regulatory events for further analysis.

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* "Fuel and Core -- Safety Restrictions" and "BWR Fuel Limits -- MCPR, MAPLHGR." These are found on pages 103 and 110, respectively, of the Cause Code List.

The first major step of this investigation was the determination of the most significant causes of regulatory loss. Within the regulatory loss category, further subdivisions may be made according to the exact cause of the outage, as listed in Section 8.1. In the United States, regulatory outages are attributed to safety limits of the technical specifications (also known as limiting conditions of operation (LCOs)), to required inspections or modifications, and to other less frequently observed causes. These different types of regulatory outages may be distinguished by sorting the database according to the regulatory information contained in the cause code for each event.

In addition to the regulatory classification provided, OPEC-2 distinguishes outages according to their urgency. The elementary categories used are "forced" and "scheduled."* For regulatory outages, the relative amounts of forced and scheduled outages serve as one indicator of the stringency or inflexibility of regulation.

The various causes of regulatory outages are listed in descending order of significance in Table 2.1 for PWRs (excluding TMI) and Table 2.2 for BWRs on

*In this work, forced outages are those that could not be postponed beyond the next weekend. Scheduled outages could be postponed beyond the weekend, but perhaps not until the next seasonal low-load period. This distinction is a simplification of that used by the OPEC-2 database.

pages 15 and 16 respectively.* The figures in each table are the average annual capacity factors lost due to regulation during 1975-1984 (in percent). Note that the majority of regulatory outages for both plant types (91.9% of PWR loss and 94.4% of BWR loss) can be attributed to the same three causes: LCO violations, inspections, and modifications.** The nature of each of these major causes is now explored in more detail.

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*Data for PWRs are reported "excluding TMI," i.e., not including the lost capacity from the two Three Mile Island plants. This distinction removes the distortion of the data due to the prolonged shutdown at these two plants. Outages at other plants brought about by regulatory directives issued in response to the accident (e.g., "TMI modifications") remain as a part of these PWR statistics.

**Note that the "Combination" category, comprising inspections or LCO violations or modifications in combination with a non-regulatory cause is included in these percentages.

TABLE 2.1: Average Annual US Regulatory Capacity Loss, 1975-1984, for PWRs, excluding TMI (in percent)
Classification by regulatory outage cause

| Outage Cause | Capacity Loss (%) |
|---|---|
| NRC-originated inspections | 3.47 |
| NRC-originated modifications | 2.33 |
| LCO violations | 1.86 |
| NRC licensing proceedings & hearings | 0.61 |
| Fuel and core safety restrictions | 0.04 |
| Combination | 0.04 |
| Unavailability of safety-related equipment | 0.03 |
| TOTAL | 8.38% |

(Combination category comprises inspections or LCO violations or modifications in combination with a non-regulatory cause.)

Source: OPEC-2 Database

TABLE 2.2: Average Annual US Regulatory Capacity Loss, 1975-1984, for BWRs (in percent) Classification by regulatory outage cause

| Outage Cause | Capacity Loss (%) |
| --- | --- |
| NRC-originated modifications | 4.45 |
| Combination | 2.86 |
| NRC-originated inspections | 1.61 |
| LCO violations | 0.86 |
| Fuel and core safety restrictions | 0.32 |
| NRC licensing proceedings & hearings | 0.16 |
| BWR fuel limits, i.e., MCPR, MAPLHGR | 0.08 |
| Unavailability of safety-related equipment | 0.02 |
| TOTAL | 10.36% |

(Combination category comprises inspections or LCO violations or modifications in combination with a non-regulatory cause.)

Source: OPEC-2 Database

### 2.2.1 Technical Specifications/LCO Violations

The limiting conditions of operation (LCOs) are one part of the technical specifications. As the name implies, they are standing safety limits that govern plant operation. In the United States, the LCOs (and the rest of the technical specification, as well) are formulated by the utility with input from the equipment vendor. Prior to initial plant start-up, the NRC must approve the technical specifications for the entire plant. Subsequently, if an LCO is exceeded at any time, the plant is legally required to shut down. In such cases where an LCO causes an outage, the OPEC-2 database attributes the outage to regulation.

### 2.2.2 Inspections

Inspections can be motivated by the NRC in two ways: 1) through surveillance requirements, also part of the technical specifications, which stipulate a certain inspection schedule for critical plant systems, and 2) through inspection/enforcement bulletins (IEBs), NRC orders requiring plants to take action, often including inspections. Note a key distinction between these two types of regulatory inspections. Surveillance requirements are standing rules for each plant, and are in effect from day to day. IEBs, in contrast, are ad hoc responses by the NRC to problems brought to its attention.

Surveillance requirements in the United States sometimes stipulate that the surveillance interval is to be variable, depending on the number of defective components encountered. As an example, suppose that a plant's pipe supports are inspected, and none are found defective. In this case, the inspection might not be repeated for one year. If one support is defective, however, the next inspection might occur in three months; if two are defective, monthly inspections might be required, and so forth.

### 2.2.3 Modifications

In the OPEC-2 database, modifications ordered by the NRC are divided into two classes: 1) modifications due to a malfunction or a construction or design deficiency, and 2) modifications due to more restrictive criteria. For simplicity, these two categories have been lumped together under "modifications," since both embrace the same corrective measure, albeit for different reasons. Some modifications are made in response to IEBs, although the majority are due to other regulatory measures.

Of particular interest during the study period 1975-1984 were the effects of the accident at Three Mile Island (TMI) in 1979 upon US nuclear safety regulation. In the two years following the accident, many inspections and modifications were motivated by the NRC through IEBs and other means. These measures were responsible for a substantial portion of the regulatory capacity loss in the years 1979 and 1980. Though the data used are not sufficient to establish a causal link between TMI and the entire increase in outages, many of the required changes addressed problems contributing to the TMI accident. In fact, many measures implemented at the plants were referred to as "TMI modifications."

## 2.3 Plant Systems Most Affected by Regulation

Table 2.3 (for PWRs, excluding TMI) on page 20 and Table 2.4 (for BWRs) on page 21 present the plant systems responsible for regulatory losses in descending order of significance. Again, the figures presented are the average annual capacity factors lost due to regulation during 1975-1984 (in percent). Steam generators are the components with the most associated losses for PWRs, while reactor coolant systems and containments are significant for both plant types. These three plant systems account for 70.6% of PWR and 79.2% of BWR regulatory losses.

Note also that economic losses are indeed small in the US statistics. No economic losses are observed for PWRs and they appear only in the twelfth rank for BWRs. This confirms the assertion made in the last chapter that the difference between capacity factor and

TABLE 2.3: Average Annual US Regulatory Capacity Loss, 1975-1984, for PWRs, excluding TMI (in percent) Classification by plant system

| Rank | Plant System | Capacity Loss (%) |
|---|---|---|
| 1 | Steam Generators | 2.74 |
| 2 | Containment System | 2.39 |
| 3 | Reactor Coolant System | 0.79 |
| 4 | Condensate/Feedwater System | 0.48 |
| 5 | Core Cooling, Safety Injection | 0.40 |
| 6 | Fuel and Core | 0.36 |
| 7 | Undefined Failure | 0.36 |
| 8 | Refueling and Maintenance | 0.23 |
| 9 | Structural/Intersystem Problems | 0.23 |
| 10 | Turbine | 0.12 |
| 11 | Chemical and Volume Control | 0.07 |
| 12 | Reactor Trip System | 0.07 |
| 13 | Electrical Systems | 0.05 |
| 14 | Circulating/Service Water | 0.04 |
| 15 | Auxiliary Systems | 0.03 |
| 16 | Condenser | 0.01 |
| 17 | Component Cooling Water | 0.01 |
| 18 | Main Steam System | <0.01 |
| 19 | Thermal Efficiency Losses | <0.01 |
| 20 | Start-up, Operator Training | <0.01 |
| 21 | Left Over | <0.01 |
| 22 | Generator | <0.01 |
| 23 | Utility Grid (Noneconomic) | <0.01 |
| | TOTAL | 8.38% |

Source: OPEC-2 Database

TABLE 2.4: Average Annual US Regulatory Capacity Loss, 1975-1984, for BWRs
(in percent)
Classification by plant system

| Rank | Plant System | Capacity Loss (%) |
|---|---|---|
| 1 | Containment System | 4.18 |
| 2 | Reactor Coolant System | 4.03 |
| 3 | Fuel and Core | 0.94 |
| 4 | Core Cooling, Safety Injection | 0.45 |
| 5 | Undefined Failure | 0.14 |
| 6 | Structural/Intersystem Problems | 0.13 |
| 7 | Circulating/Service Water | 0.13 |
| 8 | Electrical Systems | 0.09 |
| 9 | Chemical and Volume Control | 0.08 |
| 10 | Turbine | 0.07 |
| 11 | Refueling and Maintenance | 0.05 |
| 12 | Economic | 0.03 |
| 13 | Condenser | 0.01 |
| 14 | Reactor Trip System | <0.01 |
| 15 | Auxiliary Systems | <0.01 |
| 16 | Start-up, Operator Training | <0.01 |
| 17 | Condensate/Feedwater System | <0.01 |
| 18 | Main Steam System | <0.01 |
| | TOTAL | 10.36% |

Source: OPEC-2 Database

energy availability in the United States is negligible.

## 2.4 Conclusions

This chapter has established two conclusions on the nature of US regulatory losses that will be important in the rest of this investigation:

- o The regulatory instruments responsible for the majority of US loss are LCOs of the technical specifications, inspections, and modifications.

- o The plant components contributing most significantly to US regulatory losses are steam generators, reactor coolant systems, and containments.

These observations, in combination with additional information on the character of US regulatory outages, indicate an appropriate focus of inquiry for the remainder of this investigation. From here, the outage classification practices and problem management strategies of the French, Swedish, and Swiss nuclear industries may be compared with the US experience.

## 3.0 France

### 3.1 Organizations Influencing Performance and Safety

In France, three organizations make significant contributions to the safety of the nuclear industry. First, the safety regulator is the Central Service for the Safety of Nuclear Installations (SCSIN), a part of the Ministry of Industry and Research. Second, Électricité de France (EdF) is the government-owned national utility which operates all nuclear plants. Third, the Protection and Nuclear Safety Institute (IPSN) is an independent advisory body that provides expert technical support to both EdF and SCSIN. A very close working relationship is maintained among the three bodies.

#### 3.1.1 Safety Regulators

The SCSIN was created in 1973 to fulfill two duties: 1) to act as the official state advocate of nuclear power, and 2) to ensure the safety of the public and the natural environment. An organizational diagram is given in Figure 3.1 on page 24.[8] SCSIN comprises three expert groups, eight regional directors, four divisions (not shown), and a general secretariat.

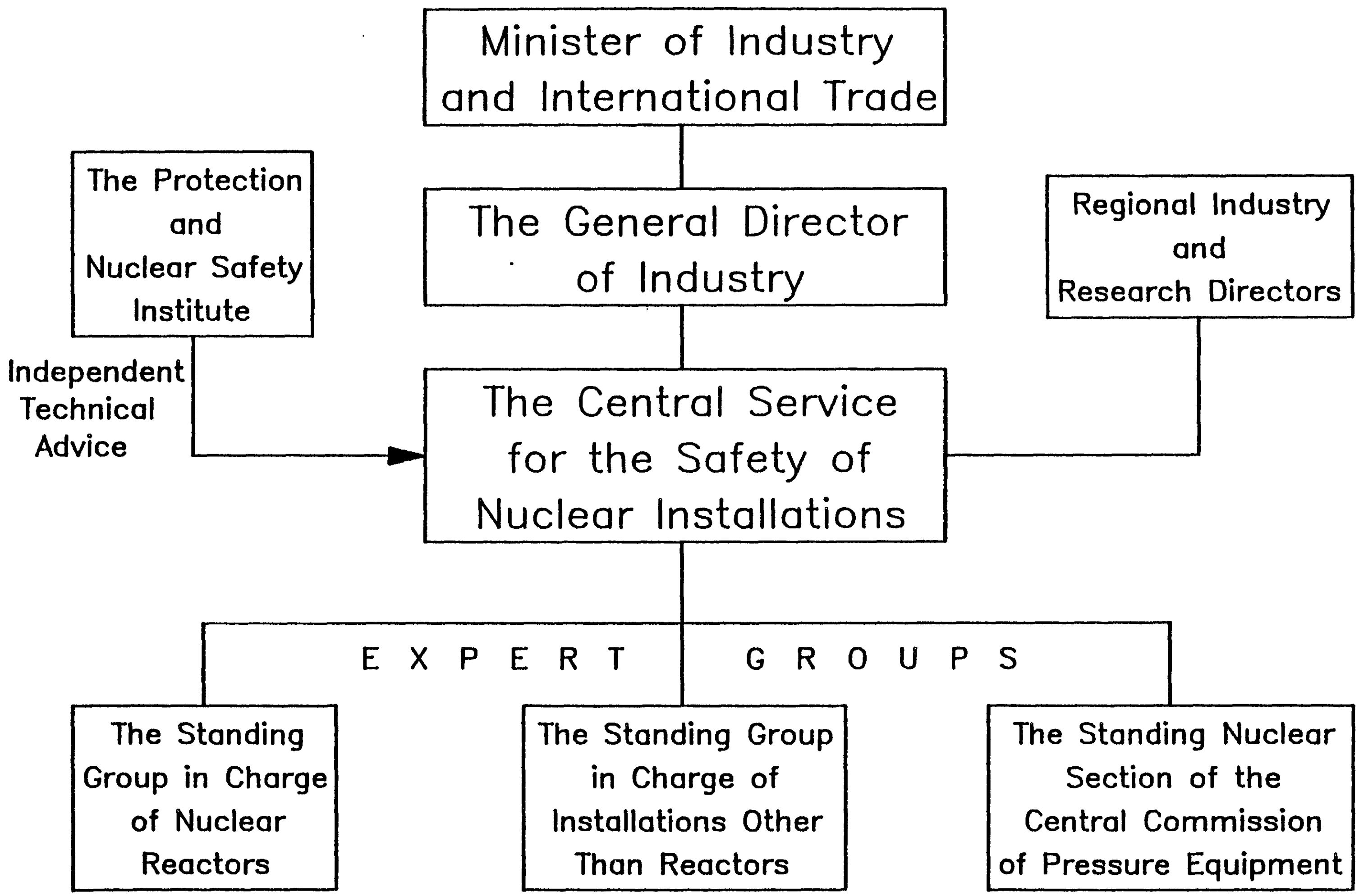


FIGURE 3.1: Organizational Structure of SCSIN
Source: Reference 8.

### 3.1.2 Utilities

Électricité de France, as the national electric utility, generates approximately 90% of France's electric power. The remainder is generated by some industries for internal consumption. Construction began on EdF's first PWR in 1969. As of 1985, 37,000 MWe of nuclear capacity were on line, amounting to 65% of France's electricity supply.

EdF maintains a substantial base of technical resources within its organization, and thus does not look to SCSIN for technical assistance. Since EdF itself is responsible for plant construction as well as operation, it is in the best position to provide for the safety of the plant, literally from the ground up. Therefore, EdF maintains a very capable technical staff to oversee all aspects of safety, in both construction and operation. At its headquarters, there is a group of technical experts that strive to identify the underlying causes of current technical problems, and to recommend appropriate action.

In addition, EdF maintains a direct technical liaison with the NRC in the United States, as well as with the regulatory bodies of other nations. EdF has access to a French database on worldwide nuclear plant outages, which is used to augment the substantial plant experience data from French plants. The OECD plant database and other resources are also available to EdF, such as those of the IAEA.

Interestingly, EdF was its own regulator for the operation of its earliest plants. The early gas reactors were built before any regulatory authority existed. The plants were actually constructed by a national engineering firm, and then operated by EdF. Such a situation illustrates the high degree of trust placed in EdF's analytical and technical capacities, which today translates to a positive and professional relationship with SCSIN.

EdF has nearly always enjoyed an excellent public image. Its twofold commitment to safety and cost containment has won the utility much support. (France has typically had the lowest cost electricity in Western Europe.) Also important, EdF is considered a prestigious place to work, and the company has no trouble attracting some of the most talented engineering graduates.

So far, EdF has not been content to rest on its laurels. Its policies and practices have continued to stress and achieve safe, economical operation. Because of the high degree of design standardization among French plants, EdF has much to gain from operating experience analysis. This program is aggressively pursued with input from operating, management, and construction personnel. Their evaluation of plant characteristics and problems helps in generating guidelines for the design of the next generation of plants.

As observed, EdF is committed to economical operating practices. Characteristically, EdF responds to problems in a plant with a "temporary fix," a remedy that safely suffices until the next refueling outage, not necessarily a repair acceptable in the long term. In this way, technical resources may be brought to bear on the problem in an unhurried, controlled, and organized way. Also, the more time-consuming, comprehensive repairs may later be conducted in conjunction with the annual refueling outage, significantly increasing the availability factor.

There are cases, of course, where the problem cannot wait until refueling, and interim measures are unsatisfactory. For the examples of valve replacement or steam generator leakage, the start of the outage may be able to be delayed a few days to a week, in order to coincide with other repair work or with a lower electricity demand period. The decision will depend on, among other things, the historical trend of the problem and the season of the year. It is important to note that such hardware problems are considered technical problems; the regulator is never blamed for any resulting outage. This point will be elaborated later.

The primary concern of most of EdF's policies is safety. As an example, consider the complete plant evaluation/inspection performed on each power station upon completion of its first year in service. This is the same comprehensive inspection that is commonly performed every ten operating years in France and elsewhere. EdF believes that no other nation's nuclear industry performs the same complete one-year evaluation.

EdF maintains an independent safety committee that is the equivalent of the Advisory Committee on Reactor Safeguards (ACRS) in the United States. The safety committee addresses current plant problems, often the same issues which the regulator, SCSIN, is studying. An example of the committee's action occurred in response to a report of ruptured tube guide pins, first from one plant, then from a second. The problem was especially vexing, because the plants were of different generations. EdF's safety committee analyzed the failures and recommended to SCSIN that staggered replacement be accomplished during refueling. SCSIN accepted the committee's proposition.

Another example of an EdF initiative came in response to problems caused by the severe winter of

1986-1987. Several plants experienced instrumentation problems due to the cold weather. These difficulties were not very severe in themselves, but they were disturbing to EdF as a possible indication that the effects of a cold winter on its plants were not well understood. Accordingly, EdF instituted a comprehensive review of the impact of cold weather on many aspects of plant operation. This action addressed not only the instrumentation problems, but was also designed to foresee and prevent other malfunctions induced by extreme weather.

To conclude this look at the nature of EdF, the utility's coherent outage management and safety philosophy is noted. Consistent with the emphasis on the technical nature of nuclear safety, EdF maintains excellent engineering resources in its own organization. Also, when technical problems cause plants to shut down, the outages are blamed on faulty equipment rather than on a capricious regulator or an unreasonably stringent specification. This emphasis on technology extends to the close, cooperative relationships with SCSIN and IPSN. In discussions with these bodies, engineers do the talking; lawyers and non-technical bureaucrats do not play pivotal roles. Finally, despite the checks and balances afforded by SCSIN oversight and the independent technical capacities of IPSN, responsibility for plant safety rests foremost with the plant operator. To date, these policies have served both EdF and the nation quite well.

### 3.1.3 Technical Advisors

The Protection and Nuclear Safety Institute (IPSN) is one of nine institutes within the French Atomic Energy Commission (CEA). An organization chart is given in Figure 3.2 on page 31.[9] IPSN has diverse responsibilities within its mandate, but for the nuclear industry, its major contributions lie in research and development, reactor safety, and radiation protection. IPSN is the main technical support for the regulatory body, SCSIN, and also has daily contacts with EdF.

IPSN was created by ministerial decree in November 1976 as a focus for CEA's efforts in radiation protection, nuclear safety, and safeguards. The organization serves the nuclear industry in particular, but also provides technical assistance to the Ministries of Industry and Research, Health, Internal Affairs and Decentralization, Transport, and Environment. IPSN employs close to 1500 people; its 1986 budget was approximately FF 1 billion.[10]

IPSN is indeed well-qualified and equipped to advise others in the field of reactor safety. Four

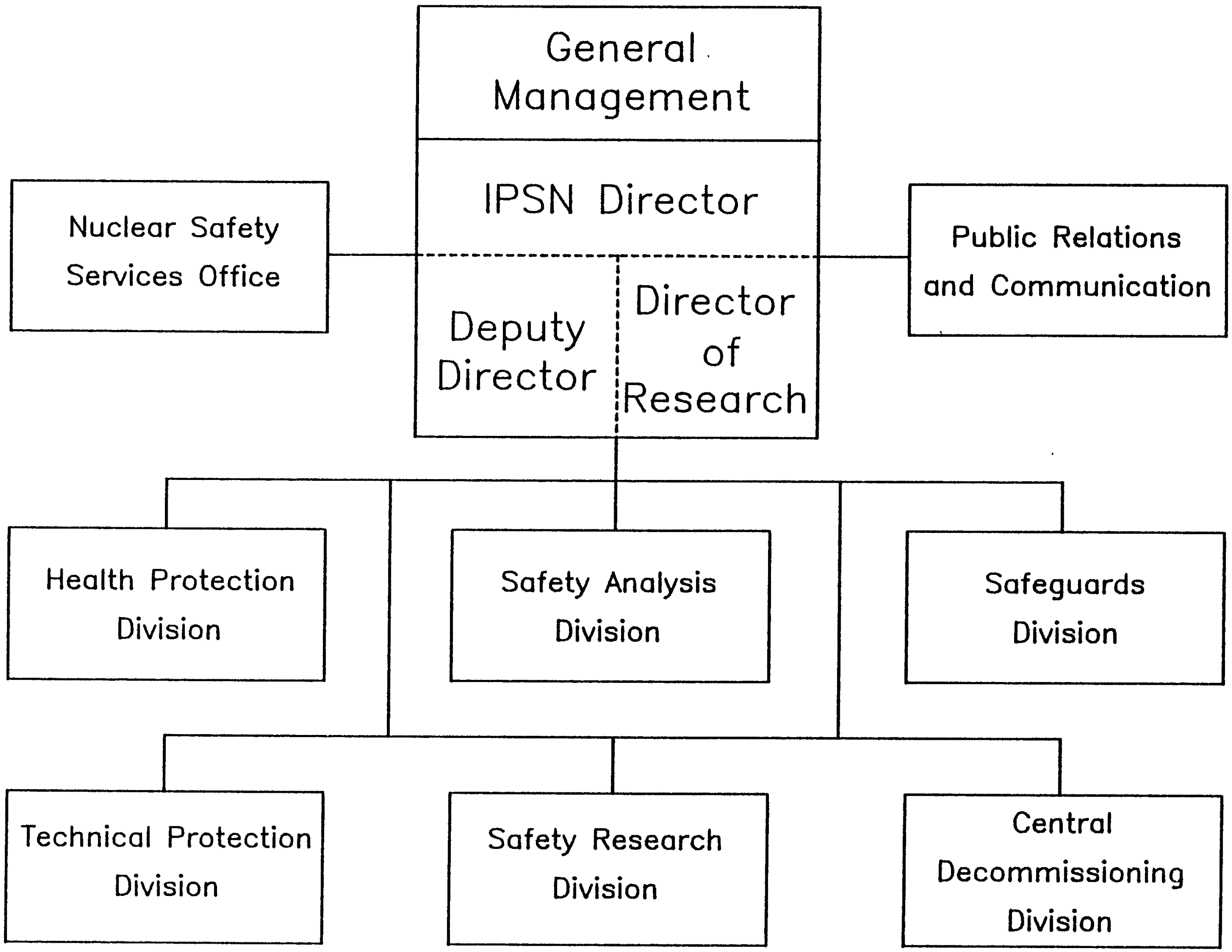


FIGURE 3.2: Organizational Structure of IPSN
Source: Reference 9.

research reactors and many experimental and test facilities (including critical facilities and test loops) are utilized by IPSN in its research programs. More than 300 technical specialists perform safety analysis studies for the Ministry of Industry and Research. These activities provide detailed and sophisticated technical knowledge that is vital for the drafting and implementation of appropriate regulations.

Nuclear safety regulation takes three different forms in France:[11]

- Ministerial orders and decrees

- Technical specifications to support ministerial recommendations

- Guidelines from component vendors

IPSN is intimately involved in preparing both technical specifications and vendor guidelines. A part of the technical specifications is referred to as the Basic Safety Rules (RFS). SCSIN prepares these rules based on the outcome of research programs conducted by IPSN specialists. The vendor guidelines include the Design and Construction Rules (RCC), which are submitted for SCSIN approval by the plant vendor. IPSN then evaluates the prudency of the proposed standards and makes a recommendation to SCSIN. The RCC, in particular, are not prescriptive guidelines. They allow plant-specific, innovative solutions, and permit plants to keep pace with technological change.

IPSN and EdF agree that plant operators are fundamentally responsible for reactor safety. Accordingly, safety analysis is chiefly based on EdF's studies and research. Operating experience data from similar plants is also useful in addressing safety questions. The high degree of plant standardization in France makes this data resource especially valuable. EdF's efforts, however, do not discourage IPSN from maintaining its own research program to check and substantiate the utility's work.

IPSN's scope of research is truly impressive. Much of its work addresses PWR power plants in particular. Included in these efforts are research on the behavior of structures and components, reliability and probability of failure analyses, and special studies of human factors considerations. This last area of interest gained additional significance after the Three Mile Island accident. Concerning risk studies, the thermohydraulics of two-phase transients in PWRs receives much attention. Other research at the various IPSN test facilities includes work on fuel behavior (including conditions of major fuel damage), the behavior of cesium and iodine aerosols under various containment conditions, and the filtered venting of containments. Finally, joint research programs are underway between IPSN and many Western nations, as well as the Soviet Union.[12]

## 3.2 Organizational Relationships in Practice

The chain of responsibility for reactor safety in France begins with each plant's operating staff. EdF as a whole has the next closest oversight, followed only then by SCSIN, the safety regulator. The regulator itself has competent technical people, and also has access to the exhaustive resources of IPSN. EdF and the individual plants, however, have the most detailed, plant-specific safety information. Hence, the technical opinion of EdF is believed and respected by the other bodies involved in the French nuclear industry. This trust facilitates a professional, constructive, and technical dialog among all parties.

There are several examples of SCSIN finding fault with EdF's procedures and standards, but even in these cases, SCSIN is not considered to have caused any plant outages that may have resulted. This is largely because of the mutual respect felt by EdF and SCSIN for their respective roles in the nuclear industry. EdF promptly informs SCSIN of any problems, and readily takes the initiative in proposing solutions, preventing the necessity of regulatory intervention. There are legal requirements for EdF to notify SCSIN immediately of unusual events at power plants; however, EdF appears to be voluntarily more forthright than required by law. After notification, EdF typically presents an informal proposal for remediation of the problem to SCSIN. This action is intended to encourage the airing of all opinions and to prepare the way for building a consensus.

Differences of opinion between EdF and SCSIN are actually rather common. A joint committee is established between the two organizations to resolve these conflicts. This group performs its function well (and usually peacefully), as no stalemates or arguments remain over the agreements forged by this committee. As a result, EdF has never been seen by the public as challenging SCSIN's policies or procedures, or as being generally unruly, argumentative, and uncooperative. Thus, the participatory process of conflict resolution in a professional atmosphere minimizes any later dissent.

An example of this process may be cited. EdF wrote a proposal to SCSIN revising the surveillance requirements for safety system testing. IPSN independently reviewed it, and some relatively minor differences of opinion surfaced among the three bodies. The collaborative committee was successful, however, in resolving the disagreement. The proposal was then accepted with minor amendments.

Private citizens, advocacy groups, and governmental bodies have never intervened substantially in the actions and decisions of EdF or SCSIN. No private individuals have ever motivated an SCSIN action, either directly or indirectly. SCSIN is not legally required to act upon or even to acknowledge any petition from the public. There have been no protracted inquiries made of EdF's affairs by its overseer in the legislature, the Parliamentary Energy Committee. This Committee is free to ask questions, but this usually occurs via informal means, such as memoranda and telephone calls rather than through formal hearings or investigations. In response to the Chernobyl accident, there were some Parliamentary discussions regarding conditions at French power plants. Some information was asked of EdF, which was supplied. Parliament seemed satisfied, as no further action was taken.

EdF and IPSN also appear to have a sound relationship. The scope of their communication encompasses technical discussions on reactor safety, and IPSN's advice and comment on EdF's proposals to SCSIN. When there is a difference of opinion between EdF and IPSN, SCSIN must choose between the recommendations, or create its own compromise. EdF and IPSN disagree fairly often in this process, so SCSIN's judgment is frequently necessary.

One avenue through which the personnel of EdF and IPSN have extensive contact is through the on-site liaison engineer from IPSN who is present during the start-up of any plant. This liaison is available for technical support and does not perform inspection duties. The purpose of having the engineer on-site is to give IPSN a firsthand knowledge of activities at the site, not to analyze the plant for possible deficiencies. IPSN considers it important to understand the depth and variety of the technical problems facing EdF, as well as the utility's solution strategies. There is, of course, much more happening during start-up preparation than one person can oversee; therefore, all activities are assigned priorities. In this way, the liaison can concentrate effort on the most critical aspects of the start-up period.

IPSN does conduct its own analysis of start-up activities. Yet, it is EdF's responsibility to recognize any difficulties and to report them promptly to IPSN so that its analysis can be performed punctually. With input from the liaison engineer, IPSN prepares a status report for SCSIN and an advisory paper for EdF. These official documents are preceded by much informal discussion amongst the three bodies. Thus, the content of the formal reports is no surprise to anyone, and rarely creates any controversy.

Another example of EdF/IPSN cooperation includes a joint research effort extending plant licensing analyses to include beyond-design-basis accidents. In 1979-1980, the two organizations adopted the practice of routinely planning for these accidents. They view such collaborative research as important, not only for enhancing cooperation but also for preventing duplication of effort. In the same spirit, even the plant vendor Framatome joins EdF and CEA in dividing and coordinating the research agenda.

## 3.3 Outage Reporting and Classification

### 3.3.1 General Principles and Examples

France reports no regulatory losses. French reactors experience the same sorts of difficulties as those in the United States, but the French industry classifies these problems differently from US industry. There is, in France, no such thing as an outage motivated by a regulator. The various requirements of SCSIN are automatically assumed to be reasonable, just, and appropriate. When some component of plant equipment violates one of SCSIN's requirements, it is the equipment that is held accountable for any resulting outage, not the requirement. According to an EdF representative, labeling outages as regulatory is deemed "unwise," and is hence not practiced.

Alternatively, human error can cause outages and thus is also cited by EdF as an outage cause. For example, some required tests are very delicate and sensitive, offering many opportunities for human error or misjudgment to cause a reactor trip. Any resulting outages are, however, attributed to human error, rather than to the set of regulations prescribing the testing.

Another cause of outages reported by EdF pertains to violations of axial offset limits. These are treated as "administrative" limits; the plant operator manages the reactor so as to stay within them. These are viewed by some in EdF as causing regulatory outages, but these outages are never reported as such.

### 3.3.2 Technical Specifications/LCO Violations

Technical specifications are written by EdF. In most cases, their content is discussed with SCSIN before their formal issuance. Thus, the final documents are no surprise to the regulator, and contain standards upon which all parties have agreed. SCSIN is free to ask questions or propose modifications to the specifications. Any such changes are effected in a joint EdF/SCSIN committee.

The baselines for establishing the first French technical specifications were modeled after those of the US reactor vendor, Westinghouse. All French plants built before 1982 were constructed by Framatome (the French nuclear vendor), using what was substantially a Westinghouse design. Even today, French technical specifications are quite similar to those for US PWRs. EdF personnel were confident that the stringency of the two nations' specifications is most often comparable, with French standards more stringent in some areas.

Technical specifications are implemented through EdF policies and observed by the plant operators. Each operator monitors trends in the critical parameters indicative of the reactor's physical state. If an LCO violation appears imminent and unavoidable by less drastic means, the operator shuts down the plant. Many times, however, the LCOs are never closely approached in operation. This is because EdF observes a set of operating specifications that are often more stringent than the LCOs contained in the technical specifications. EdF adopts these conservative policies in the interest of achieving plant lifetimes of 30-40 years. In order to realize reliable operation over this period of time, EdF believes that both operating and maintenance practices must be painstaking and exacting, erring only on the side of conservatism and prudency.

One notable case may be cited of a disagreement between EdF and SCSIN over the implementation of technical specifications. In a power plant quite close to the French border with the Federal Republic of Germany, SCSIN ordered a shutdown before any LCO was closely approached. This was done due to the sensitive location of the plant. EdF disagreed with the order, but did not delay in obeying the directive. This outage was considered a technical specification violation, as SCSIN's judgment is never blamed for an outage. Thus, SCSIN merely acts to call attention to objective conditions in the plant. Such conditions, when evaluated using standards and criteria accepted by EdF and SCSIN, may necessitate a shutdown.

### 3.3.3 Inspections

During regular plant operation, no inspector is present. Frequent plant visits are preferred instead, customarily two per month. The date of the inspection and the agenda for the visit are established in advance to ensure that the plant staff and other technical resources are available for discussions and analysis.

Nearly all routine inspections in France are accomplished in conjunction with other outages, particularly the annual refueling outage. To appreciate the success of French inspection policies, note that the equivalent energy availability lost in 1986 due to routine inspections of French 900 MW PWRs was only 0.08%. For comparison, the average value (1975-1984) of annual capacity loss due to so-called "regulatory" inspections for US PWRs (excepting TMI) was 3.47%, over 43 times greater.

One explanation for the discrepancy in inspection outages lies in the nature of the two nations' surveillance requirements. Unlike the United States, France does not have a variable surveillance interval that depends on the number of component malfunctions. Furthermore, France has nothing analogous to the US inspection/enforcement bulletins. Yet, despite an apparently more formidable set of US regulatory requirements, standards demanded in French plants are no lower. For example, recall that a comprehensive ten-year inspection is conducted in each French plant after only one year of operation. This inspection is conducted as a matter of policy by EdF; it is not the result of any regulatory directive.

### 3.3.4 Modifications

France did not make as many modifications in response to the TMI accident as did some other nations. Some modifications that were typical reactions to TMI in other countries had fortuitously been effected by France prior to the accident. For those changes implemented in France after the accident, a concerted effort was made by EdF and SCSIN to schedule the work during annual refueling outages. In general, EdF devotes many resources to outage planning and scheduling, to minimize forced outages and extensions of planned outages. Many of EdF's post-TMI modifications addressed human factors concerns in the control room.

EdF representatives could not recall any post-TMI modifications that were too urgent to be delayed until refueling. There were a few plants where cracks were discovered in steam generator outlet piping, but this was probably not TMI-motivated, according to EdF. With certain time restrictions, SCSIN allowed EdF to devise its own schedule for remediation. In some cases, the work necessitated shutdowns prior to refueling. Nonetheless, the outage cause reported was pipe cracking rather than a regulatory order. It is likely that 1) the high degree of standardization among French plants, and 2) the existence of only one electric utility, EdF, were significant reasons why

SCSIN allowed EdF such discretion and freedom in scheduling repairs. Most any problem is likely to be generic in the plants, as all share the same types of components. Furthermore, EdF's standards and policies apply to all French plants; these standards are well-understood by SCSIN.

## 3.4 Comparisons with US Experience

As noted in the section on technical specifications, EdF observes a set of operating specifications more stringent than the officially established LCOs. These company standards are adopted voluntarily, without regulatory pressure; EdF considers them one element of prudent engineering practice. This is not the case in the United States.

Regarding inspections, France has no resident inspectors at the plants, no variable surveillance intervals, and nothing analogous to an IEB, unlike the United States. Yet, between the efforts of EdF and SCSIN, French inspection requirements are similar to those in the United States. EdF appears to compensate for the less demanding regulatory surveillance with its own policies. For example, performing the comprehensive ten-year inspection after only one year of operation is an EdF practice not seen in the United States.

SCSIN never forced a modification outage upon EdF. At most, SCSIN's requested changes were discussed with the utility, and implemented by EdF to the regulator's satisfaction. Indeed, no regulatory impositions have ever been made in France; conflicts between the regulator and utility are always resolved via technical discussions. In the United States, technology frequently plays a muted role in such dialogs, where legal concerns dominate the agenda.

It is apparent that while French regulations are in some areas less intrusive than US statutes, EdF makes up the difference with its voluntary safety practices. As observed, these are frequently more stringent than the regulation itself. Thus it is usually EdF's policies that are responsible for outages. Even if an SCSIN requirement should reach beyond EdF's standards, any consequent outage is considered a technical difficulty and never a regulatory imposition.

Four fundamental differences between the US and French nuclear industries should be borne in mind when making regulatory comparisons:

- Plant age. French plants (indeed, those in all of Western Europe) are significantly younger, on average, than US plants. Many of the most troublesome problems (steam generator leaks, for instance) worsen with age.

- Plant standardization. With the tremendous degree of standardization observed in France, data from the detailed inspection of one plant may be statistically extrapolated to all other plants, barring plant-specific features. Hence, detailed inspections may be distributed among all the plants. Furthermore, standardization implies that most every problem is generic. The resources of EdF may thus be concentrated on these difficulties (relatively few, yet probably widespread) rather than on a host of local problems at each plant.

- Utility diversity. EdF is a monolith. Its policies are well-known to SCSIN and to IPSN. Furthermore, these policies are applied consistently to all plants (this is partly facilitated by standardization, in turn). Thus, with a minimum of effort and inquiries, SCSIN may be confident of understanding how any problem at any plant will be attacked. The same may not be said of the NRC's position in the United States.

- Utility size. EdF's size is an asset in achieving and maintaining a high level of technical competence. In the United States, there are dozens of nuclear utilities with varying sizes and technical capacities. Some small utilities do have excellent technical resources, but in general, small size and meager budgets are tough obstacles to establishing and maintaining high-caliber technical abilities.

There remain significant differences between the industries of the two nations, however, that are not plausibly explained by the factors above. For instance, the many positive characteristics of the EdF/SCSIN relationship in France form a stark contrast with the nature of the typical utility/NRC relationship in the United States. In the French industry, a professional, mutually respectful, and technically oriented environment prevails. In the United States, the atmosphere is instead characterized by mistrust, litigation, and poor plant performance.

The French utility, EdF, is technically excellent in its own right, independent of SCSIN or IPSN. The utility is treated as a technical peer of the regulator or IPSN. Because of this common level of competence, a permanent technical dialog exists among the three, one that evolves to address new issues as they arise. Again, the US situation is sharply different. As many utilities have only mediocre technical abilities, the NRC is relied upon for the analysis and assurance of reactor safety. The NRC also houses regulatory and technical analysis functions under one roof. In

France, these roles are filled by separate organizations, SCSIN and IPSN respectively. Technical analysis is much less likely to be tainted by nontechnical factors when the two functions are separated.

IPSN and SCSIN officials shared opinions of the US nuclear industry, utilities and NRC alike, in relation to that of France. Regarding utilities, several representatives perceived that the US utilities were generally more reticent than EdF to accept responsibility for the technical state of their plants. Additionally, utility management in the United States was thought to suffer from inadequate organization. Regarding outage reporting practices, one EdF official felt that US utilities readily blamed excessively stringent regulation for what would be considered technical problems in France. Addressing and avoiding such technical problems at the plants would be simply part and parcel of EdF's voluntary engineering practices.

Concerning the NRC, EdF representatives perceived US regulation as "going too far," or being too prescriptive. It was also recognized that greater prescription may be required (or at least be more convenient) when dealing with nonstandard plants operated by a wide variety of utilities. Whatever technical expertise that does exist at nuclear plants, however, is rarely sought by the NRC. Lastly, the tradition of the resident inspector at US plants epitomizes the mistrust between regulator and utility, and points to the dearth of a professional relationship between them.

## 4.0 Sweden

### 4.1 Organizations Influencing Performance and Safety

The Swedish Nuclear Power Inspectorate (SKI) is Sweden's nuclear safety regulatory authority, analogous to the NRC in the United States. Other administrative and scientific bodies in Sweden are responsible for related nuclear safety issues, namely:[13]

- National Institute of Radiation Protection (SSI). Established by the Radiation Protection Act, SSI works closely with SKI to ensure safe conditions at all nuclear facilities.

- Swedish Plant Inspectorate (SA). Conducts testing of pressure vessels at nuclear installations.

- National Board for Spent Nuclear Fuel (NAK). Oversees technical research and coordinates financing for the handling and disposal of spent fuel.

- National Environmental Protection Board (SNV). Created by the Environment Protection Act, SNV monitors non-nuclear disturbances in the vicinity of nuclear facilities.

- National Board of Occupational Safety and Health, National Electrical Inspectorate. These bodies exercise the same surveillance duties at nuclear and non-nuclear installations.

In addition, each of the four municipalities having a nuclear power station nearby maintains a local safety committee. These groups keep informed of current and proposed nuclear safety and radiation protection regulations at their respective power plants. The committees also perform public information and emergency planning duties.

### 4.1.1 Safety Regulators

The principal functions of SKI, the safety regulator, are nuclear facility licensing and oversight, the promotion of safety, and the supervision of handling and storage of fissionable nuclear material. SKI and the utilities both stress, however, that primary and direct responsibility for plant safety lies with the utility operating the plant. Supplementary duties of SKI include coordinating technical research and development in nuclear safety and communicating nuclear safety information to the public.

An organization chart is given for SKI in Figure 4.1 on page 52.[14] The SKI Board is appointed by the government; the Director General serves as its chair. Reporting to the Board are the two technical Offices,

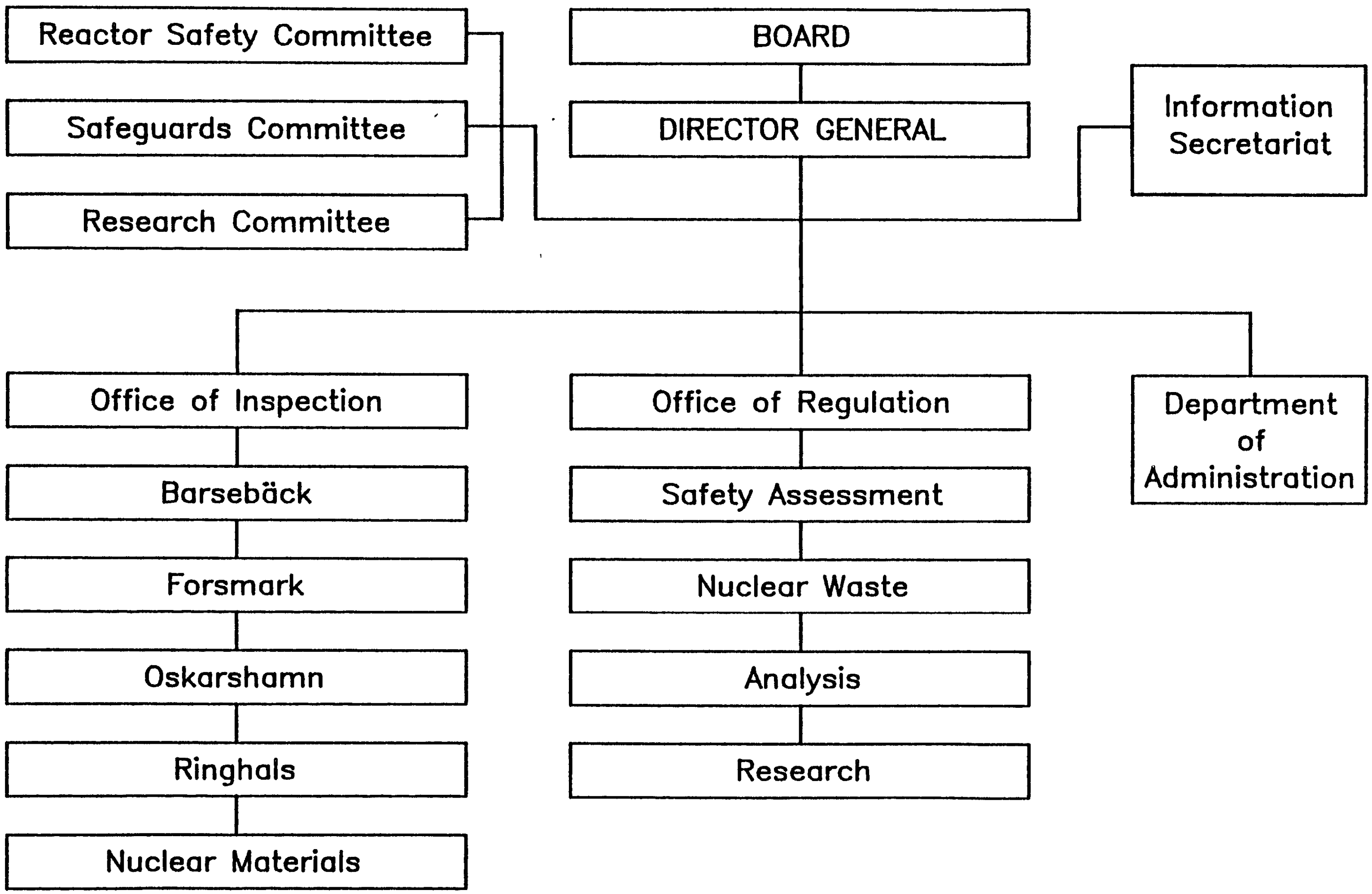


FIGURE 4.1: Organizational Structure of SKI
Source: Reference 14.

the Office of Inspection and the Office of Regulation, as well as the Administration and Information sections. The key functional distinction between the two offices is that the Office of Regulation is responsible for the text and specifications of regulations, while the Office of Inspection is concerned with enforcement of these regulations. The Inspection Office assigns one liaison inspector for each plant. This official is not an inspector in residence at the site (there are none); instead, frequent visits are made.

Also indicated in the chart, three advisory committees are a part of SKI:[15]

- The Reactor Safety Committee keeps informed of SKI's supervisory activities and provides technical advice on reactor safety and licensing matters.

- The Safeguards Committee advises SKI on safeguarding nuclear material, including measures to combat theft and sabotage committed against a nuclear facility or a transport vehicle.

- The Research Committee proposes and evaluates research projects and is available as an advisor to the Research Division.

SKI is financed with monies from the nuclear power utilities and with funds allocated by the government. The total budget for budget year 1984/85 was SEK 25,000,000 (roughly $3 million at that time). An additional SEK 44,400,000 (approximately $5 million at that time) was allocated for nuclear safety research.[16]

SKI currently has about 85 employees: 60 professionals and 25 supporting personnel. All employees work at a single location in Stockholm.

#### 4.1.2 Utilities

Sweden has four utilities owning nuclear power plants: the state-owned Swedish State Power Board, and three (at least partly) private utilities, Forsmarks Kraftgrupp AB, Sydkraft AB, and OKG Aktiebolag. In 1980, partly in response to the Three Mile Island accident, the four nuclear utilities formed the Nuclear Safety Board of the Swedish Utilities (RKS). As of 1987, RKS merged with the personnel training organization run by the utilities to form the Nuclear Training and Safety Center (KSU), which today houses the utilities' safety collaboration efforts and training programs.

KSU both coordinates the in-house safety efforts of the individual utilities and conducts its own safety projects, drawing on the combined resources of the utilities. Much of KSU's attention is devoted to managing the experience feedback program, whereby operating data from domestic and foreign plants are collected, analyzed, and disseminated in useful form to the nuclear utilities. Experience feedback aims to provide each plant with a relevant set of data on technical disturbances that can serve as an information resource in anticipating and resolving problems. In addition, KSU and the utilities emphasize outage planning and investment in high quality, high availability measures.

## 4.2 Organizational Relationships in Practice

In discussions with SKI, the safety regulator, and KSU, the utilities' own safety organization, both groups characterized the regulator/utility relationship as positive, technically oriented, and cooperative rather than adversarial. The two bodies seem jointly committed to a safe and viable nuclear industry in Sweden. To this end, SKI takes pains to give utilities the maximum possible advance notice of upcoming regulatory actions, so that each plant may schedule outages most efficiently.

There was no instance of a regulatory order so extreme that it forced a plant to cold shutdown. KSU noted, however, that plant start-up from a shutdown (usually refueling) had sometimes been delayed by regulatory action (see next paragraph). KSU was asked how SKI was likely to implement changes in the stringency of regulations, for example, modifications arising from revisions in computer codes for seismic analysis. KSU felt that the utilities would be allowed a reasonably long time (i.e., until the next refueling) within which to schedule the necessary work. This contrasts with the NRC's handling of the issue of seismic design criteria, revisions in which were sufficient cause for near-immediate shutdowns.

In the opinion of KSU officials, the most extreme regulatory action taken by SKI occurred upon the discovery of cracked piping at the Ringhals 1 plant in August 1986. This defect was uncovered in the last week of the refueling outage. As a result, Ringhals 1 was down for four more weeks for inspection and welding work. SKI also issued a statement giving notice that all BWRs would be inspected within a short period of time.

Another anecdote illustrates the usual outcome when there is a technical difference of opinion between a utility and SKI. In the Oskarshamn 3 and Forsmark 3 plants, the main steam isolation valves had been newly repaired. Inspections of the valves were to occur at intervals ranging from weekly to every three weeks. The respective utilities felt this inspection frequency to be unnecessarily conservative; they requested a change in inspection frequency, decreasing it to every four weeks. SKI did not permit this change. At other times, ring inspections in generators and requirements for auxiliary power supplies have elicited utility/regulator disagreements.

In all these cases, the utility concerned expressed its dissent in numerous discussions, but eventually complied with SKI's wishes. Vehement protests by utilities over safety regulation do not occur in Sweden, as it would be very damaging politically to be seen arguing with SKI. With the current moratorium in Sweden (see p. 67), as well as post-Chernobyl anxieties, the industry's position is already tenuous at best. Thus, the precariousness of nuclear power can create a rather placid regulatory environment.

The availability of new analysis techniques enabling more sensitive monitoring or more accurate modeling have not forced the regulator's hand, causing SKI to shut down plants until the new methods may be implemented. In reality, many new techniques are developed voluntarily by the utilities themselves. This work is, as one might expect, enthusiastically encouraged by SKI, but SKI has never forced new methods on a utility. With the foresight and initiative generally demonstrated by the utilities, SKI may never have had the opportunity to do so.

Indeed, the Swedish utilities frequently lead SKI in recognizing and addressing technical problems. The individual plants do seem justified in maintaining (and SKI agrees) that each plant is its own technical expert, and that primary safety responsibility lies with the operator. There are strong incentives for the utilities to be alert, competent, and cooperative. Just as it is unwise to appear an adversary of SKI, likewise, the utilities cannot afford to let SKI lead them around, scolding and cajoling them, presenting the regulatory hoops through which the utilities must jump. One of the KSU staff observed that the typical utility/regulator relationship in Sweden was characterized by "more dialog than directives."

Representatives of KSU were asked if there was ever any complaint from SKI of untimely notification of the regulatory authority regarding events at a plant. No significant cases of this were recalled. The reporting routine followed in Sweden includes daily operating information transmitted to SKI from each nuclear utility. Any deviation from normal conditions is to be reported; requirements governing the utilities' response to most such deviations are incorporated in the standard operating procedures of each plant. SKI does have the authority to require that extreme measures be taken (including plant shutdown), but as noted above, such extraordinary cases have not occurred.

## 4.3 Outage Reporting and Classification

### 4.3.1 General Principles and Examples

Sweden reports few regulatory losses. Nationally, the highest regulatory capacity loss figures were about 12% for PWRs (in the years 1975, 1982, and 1983) and 4% for BWRs (in 1976).[17] All other years had an insignificant amount of these losses. Reactor technology is quite similar in the United States and Sweden; one would expect the same technical problems in the two nations, but perhaps different criteria for classifying the resulting outages as regulatory or otherwise. Thus, it was necessary to ascertain what does and does not constitute a regulatory loss in Sweden.

In a discussion with KSU, its representatives indicated that required redesign of steam generator preheater sections has been considered a regulator-imposed loss. This activity was responsible for the high capacity losses in 1982 and 1983. One explanation given for the absence of many large losses was that most of the tasks required by the regulator could be accomplished during the annual refueling outage. There is a cooperative effort between SKI and the utilities to schedule nearly all regulatory work during this time. Such was the case with modifications due to hydrodynamic load calculations for BWRs. Another example of a regulatory outage occurred at the Oskarshamn I plant. A recent refueling period was extended for replacement of the core grid when the grid support bolts cracked. Yet another case of a regulatory loss occurred at Ringhals 2 with a charging pump problem.

Noting all these examples, one of the KSU representatives offered a general definition of a regulatory loss in Sweden. He proposed that nearly all such losses arise from an unresolved technical difference of opinion between SKI and a utility. If SKI insisted on its viewpoint in these situations, the utility was very likely to label any resulting outage as regulatory. Most of those present accepted this statement, although some preferred a more inclusive definition.

As mentioned above, one reason for the infrequency of SKI-motivated losses is that much of the work that SKI might require is scheduled in conjunction with other outages, particularly refueling. Utilities maintain a "stop list" of pending repairs and preventive maintenance, which assigns priority to these tasks for the next forced or scheduled outage. Thus, unless SKI's required work forces the shutdown or significantly extends an outage, the outage is not considered regulatory.

As an example of an outage (a derating, actually) not considered regulatory, consider the Ringhals 2 plant. It has been operating at 80% power since last year when cracks were found in the steam generator tubes. The steam generators had already been scheduled for replacement in 1989, however, so the utility was interested in prolonging their life until that time. The utility found (and SKI agreed) that by lowering the plant's power just 20%, pressure and temperature would drop such that crack growth essentially stopped. SKI probably would have prohibited operation if the cracks continued to grow, but it was essentially the utility's decision to derate the plant. Thus, this derating was not treated as a regulatory imposition. It was mentioned that, in the case of an SKI-ordered derating, the lower power level is adopted as a new baseline for calculating capacity losses. This practice masks these deratings in the data, so these must be treated independently when seeking regulatory losses.

At the Ringhals 1 plant, some four-inch piping was found to have intergranular stress corrosion cracking (IGSCC). This might have resulted in a regulatory loss, but no shutdown was required. The repairs were accomplished while operating. In the wake of this discovery, no other plants were forced down for inspection or testing. There are numerous cases in the United States where problems at one plant have prompted the NRC to call for inspections at others, often necessitating shutdowns.

Next, the stringency of different facets of Swedish safety regulation will be evaluated: technical specifications, inspections, and modifications.

### 4.3.2 Technical Specifications/LCO Violations

Swedish technical specifications are drafted by the utility, drawing upon the data and experience of the plant's vendor. The specifications must be approved by SKI. A joint committee is convened to resolve significant differences of opinion prior to official approval. In this way, there is much informal contact and negotiation between utility and regulator. These processes make the official approval of the specifications by SKI very straightforward. Some questions of interpretation and understanding may remain, but these are normally inconsequential for the plant's operation.

Swedish technical specifications come in two varieties. There are specifications for standard operation, and there is also a special set of specifications that apply during refueling outages. In addition, there are rules and procedures separate from either set of technical specifications that are embodied in the Surveillance Test Book (STB). The STB contains requirements that are more detailed and plant-specific than the technical specifications. Criteria for the performance of individual plant components are enumerated in this document. The STB is not ancillary to the technical specifications; both documents carry equal weight and standing. Like the technical specifications, the STB can be revised and reworked by a joint SKI/utility committee.

Technical specifications are applied rather conservatively in practice. Utilities customarily monitor critical parameters of the reactor's operation to ensure that the plant remains safely within the LCOs. If a trend in one of these parameters indicates an impending LCO violation, the utility itself will shut the plant down if no lesser corrective action is effective. Alternatively, SKI may see that the plant is already in violation of the LCO and order a shutdown. The latter is much less frequent, as the utilities view it as good practice (and in their own best interest) to remain within technical specifications and within the strictures of the STB. Such utility-motivated shutdowns are not classified as regulatory losses, whereas the rare SKI imposition is considered a regulatory loss.

KSU representatives were asked if violations of technical specifications are clear-cut, deterministic events, or rather, if judgment and discretion are often applied in establishing violations. They responded that some negotiation occurs over whether or not a plant would shut down when in violation of specifications. Usually, however, there is not much discussion, as the specifications include restrictions on the time allowed until shutdown or repair that are observed without question.

Technical specifications may be changed via a process analogous to the process of their initial formation. The utility desiring the change proposes a new text which then must be approved by SKI. Such changes are actually quite common, especially in the early years of a plant's operation. It is noteworthy that the only participants in the discussion of proposed revisions are SKI and the utility, i.e., no public hearings are held. Swedish citizens must go directly to the government with grievances; this avenue of dissent has not been used frequently.

### 4.3.3 Inspections

SKI does not employ resident inspectors at the plants. Creating such a position might be considered an anti-cooperative gesture by KSU; indeed, some KSU personnel saw such inspectors in the United States as a manifestation of the distrust between the NRC and the US utilities. At the Forsmark plants in Sweden, the SKI inspector visits about every two weeks. The visit is customarily announced in advance; surprise inspections are a rarity. According to KSU, a "continuous dialog" exists between each utility and its SKI inspector. SKI itself has daily contact with these inspectors.

Plant components are inspected according to a constantly evolving hierarchy of emphasis. Those systems exhibiting the most problems will receive the greatest amount of scrutiny. It appears that the amount of surveillance and testing done in Sweden and the United States is commensurate, with Sweden performing occasionally more. The major difference is not the amount of testing and inspection, but rather how systems are selected for testing in each country.

### 4.3.4 Modifications

KSU personnel could recall no modifications required due to construction or design deficiencies.

SKI has, however, issued more restrictive criteria which required plant modifications. The most notable example of this action was in response to the TMI accident in the United States.

SKI devised an "action program" (with some input from the utilities) in the wake of the TMI incident comprising certain required backfits. In 1979, Ringhals 2 was the only PWR in operation; it was actually conducting refueling at the time of the accident. SKI's required modifications extended the refueling outage for three weeks. In addition, SKI added mandatory safety reporting requirements for all nuclear plants. Periodically, an as-operated safety analysis report would be required, as well as a comprehensive ten-year safety report after each decade of operation.

A second major reaction to TMI came from the government, although SKI was also involved. A commission was assembled to conduct a reactor safety study, which was completed in early 1980. The utilities, universities, and SKI were among the collaborators. This study was one component of widespread discussions on the safety of nuclear power after TMI. Because of the attention now focused on the industry, the construction of the Forsmark plants 1, 2, and 3, Oskarshamn 3, and Ringhals 3 and 4 was delayed at least one year.

In March 1980, shortly after the reactor safety study was released, a public referendum was held on the future of nuclear power in Sweden. People voted in favor of phasing out nuclear power completely by 2010. KSU representatives felt that the TMI accident was the primary motivator of the referendum and the major determinant of its outcome.

## 4.4 Comparisons with US Experience

The discrepancy in the amount of regulatory loss reported by Sweden and the United States is due almost entirely to the outage classification scheme used by each nation. Some differences exist in the means of regulation; for instance, Sweden has no resident inspectors and nothing analogous to an IEB. However, similar technical specifications and inspection requirements are in place in the two nations. Where differences appear, Swedish regulations are usually stricter, as with their conservative application of the LCOs. More information on technical specifications is given below. Furthermore, the same technical problems were most troublesome for both countries. The key difference is that when outages are incurred (whether for technical specification violations or for inspections), Swedish utilities hold the hardware accountable, labeling the difficulty as a technical problem. In the United States, it is the regulation or the regulator that is considered the cause of the outage. Hence, far more regulatory outages are reported in US performance statistics than in those of Sweden.

Comparing technical specifications in Sweden with those in the United States, some differences appear. Efforts are made in Sweden to create specifications that are site-specific and plant-specific. US specifications tend to be more generic. For the early Swedish plant specifications (drafted in 1973-1974), the baseline data was taken from US vendor information (i.e., from Westinghouse). Specifications used by ASEA-ATOM, the Swedish vendor of most later plants, are somewhat more stringent than those in the United States. In particular, BWR specifications are more demanding concerning water chemistry requirements. In addition, Swedish technical specifications stress functional requirements of plant systems, e.g., leakage rates, rather than prescriptive standards, e.g., specifications on pipe or vessel integrity.

Representatives of KSU were certain that US regulations were far more voluminous than those of Sweden. This observation must be interpreted with care, because any difference in regulatory scope may be due to significant differences between Swedish and US utilities. On balance, Swedish utilities are more technically competent, take greater initiative with safety measures, and set higher internal standards than do US utilities. Hence, a less prescriptive and less intrusive set of regulations may be justified for more vigilant utilities, as those in Sweden appear to be.

Unique among the European nations studied, Sweden did report some regulatory loss. It was associated with the redesign of steam generator preheater sections. This was classified as regulatory because there was an unresolved technical difference of opinion between SKI and the utilities over the shutdowns.

## 5.0 Switzerland

### 5.1 Organizations Influencing Performance and Safety

#### 5.1.1 Safety Regulators

The nuclear safety regulator in Switzerland is the Swiss Federal Nuclear Safety Inspectorate (HSK), within the Swiss Federal Office of Energy. Broadly, its responsibilities are nuclear safety and radiation protection. Similar to the NRC, the major technical functions performed by HSK are conducting safety analyses for proposed power plants, writing technical reviews, and performing surveillance and backfitting analyses for existing plants. Unlike the NRC, however, HSK has no legal section. This is so because HSK operates in a less politically charged environment. A separate section of the Office of Energy assumes legal responsibilities, such as the granting of plant permits. This section is also the legal advisor for HSK.

The Office of Energy in the Federal government grants three different permits to nuclear plants: general, construction, and operating. HSK's responsibility includes any decision for a plant within the purview of one of the three permits. As an example, when the Swiss seismic risk charts were developed in 1979, all plants had to be requalified.

In particular, the anchoring of electrical panels was analyzed and modified if necessary. This action required no change in the operating permit, so HSK managed the recertification.

Currently, HSK employs a total of 54 people, professional and supporting. The 1983 budget (latest figures available) was approximately SF 5 million. An additional SF 10 million was allocated for external research contracts. See Figure 5.1 on page 72 for a diagram of the structure of HSK.[18]

The Swiss Corporation for Pressure Vessel Supervision (SVDB) has authority over pressure vessel quality. The Corporation has conventional and nuclear divisions; the nuclear section works under HSK. No resident inspector is present at the plants, only a part-time liaison who performs periodic checks. There has been some support from the government for appointing a full-time resident inspector at each plant.

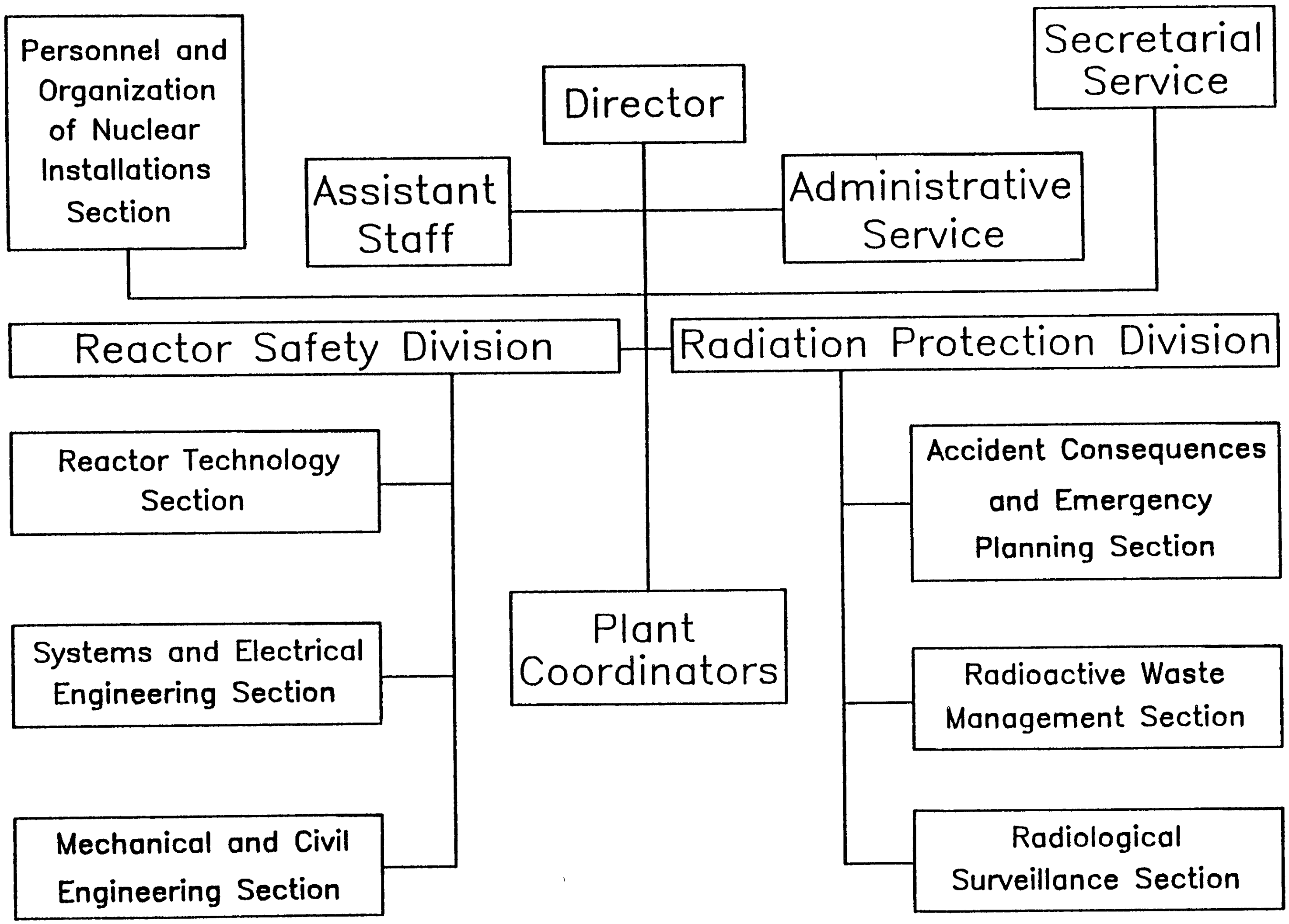


FIGURE 5.1: Organizational Structure of HSK
Source: Reference 18.

#### 5.1.2 Utilities

The Swiss nuclear utilities employ rather small staffs. The technical departments are small as well, but highly dedicated and competent. Each plant maintains a skilled engineering staff on-site; there is no separate technical headquarters. There is much emphasis on hiring quality workers, whether engineers or laborers, and on doing so early in the start-up phase. This practice, in combination with a turnover rate of only a few percent each year, creates a dedicated and professional staff thoroughly familiar with the plant.

The utilities emphasize quality and redundancy in the entire plant, in both the nuclear and balance of plant fractions. HSK supports this policy. The utilities feel that non-nuclear induced stresses on a plant (scrams caused by turbine trips, for example) are just as severe as many nuclear events, from a safety standpoint. Hence, pains are taken to avoid such events; quality assurance measures and redundancy permeate the entire plant. This is not tantamount to introducing greater complexity into the plants. On the contrary, utilities strive to keep the plants simple and transparent.

While quality is a priority, the utilities manage the plants to maximize their availability according to:

Availability = F(Reliability, Maintenance)

This equation is applied to safety equipment as well as to the power production system. The relation implies that high quality (reliability) measures requiring scheduled outage time are adopted to decrease the amount of forced outages (maintenance) experienced. This is done up to the point that the forced outage time saved equals the scheduled outage time invested. The mathematical relation is always tempered by safety considerations; reliability is felt to be intimately connected with safety.

In planning scheduled outages, the utilities budget the time to prevent hurried maintenance. Efforts are made not to overwork the employees. This keeps the staff alert, and minimizes carelessness. Also, time is reserved for unanticipated work during the outage, which always seems to arise.

## 5.2 Organizational Relationships in Practice

The professional staffs of both HSK and the utilities employ engineers and scientists exclusively. Thus, their discussions are always technically rather than legally oriented. The dialog is "fair, open, and tough."

Even in the relatively small Swiss nuclear industry, considerable diversity exists. Out of the five nuclear utilities which own at least a share of a nuclear plant, a range of technical opinions is usually found. By themselves, the utilities have formed a national association to discuss technical aspects of reactor safety and to promote excellence. Their differences of opinion are often lessened through these meetings. As the regulator, HSK adopts a more conservative position than most utilities, but the gap is rarely large. Diversity is also found in the variety of Swiss nuclear plant designs. Excepting possibly the two Beznau plants, no two of the five plants operating in Switzerland are markedly similar.

Despite such diversity in both utilities and plant hardware, HSK refrains from drafting prescriptive requirements. Rather, HSK's standards are functionally oriented, requiring a certain level of performance instead of specifying the use of certain technologies and systems. Furthermore, HSK strives to be consistent in interacting with all the nation's utilities. Since HSK does not divide the industry into regions, geographically or otherwise, most of HSK's officials are well-acquainted with key personnel at each plant. Of course, the small size of the Swiss nuclear industry significantly reduces the need for administrative partitioning or layering.

Such policies of HSK have created a peaceful and cooperative environment for the nuclear industry in Switzerland. Nevertheless, opportunities for the resolution of disagreements and the amendment of requirements have been provided by policy and law. For example, if a utility believes that a certain inspection requirement is no longer reasonable, it may propose and defend a new standard in meetings with HSK. The final decision on such a proposal, however, is HSK's alone. Even if the proposal is rejected by HSK, there remains some latitude for negotiation on the exact timing of any outages that may be required. Swiss utilities have availed themselves of this proposal and review process numerous times, often with success. This demonstrates that HSK and utility perceptions of safe and prudent industry practice are largely in agreement.

Before passing judgment on a controversial issue, HSK is required to hear the arguments of the utility. If a utility is not satisfied with HSK's subsequent findings, it may appeal to the Federal government for another decision. None, however, have ever resorted to this avenue of redress.

The harmonious professional relationship between the regulator and each utility is created and supported by their close and regular communication. Each plant is required to submit a monthly operating report to HSK. Moreover, unusual events are to be reported immediately, so that HSK is apprised of the plants' condition. Potentially dangerous conditions at plants close to international borders also entail notification of other nations. Also of importance are periodic meetings between HSK and the staff of each plant to discuss both general concerns and plant-specific matters.

The Swiss response to the widely experienced problems of escalating plant costs and extended construction periods serves as an excellent example of industry-wide collaboration. In Switzerland as around the world, there was great interest in containing plant capital costs and shortening construction times without compromising safety or availability. To this end, a task force comprising representatives of plant suppliers, engineering firms, and the regulatory authorities worked on these issues under the sponsorship of the Federal Office of Energy. The task force met frequently from 1983 to 1986. The final report* issued in January 1986 presented many promising recommendations for reducing costs and construction times, and for backfitting existing plants.

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*The report was entitled Projektabwicklung und Qualitätssicherung bei Kernkraftwerken (PQS), or Project Management and Quality Assurance for Nuclear Power Plants.

Inspection practices in Switzerland reflect the same collaborative, cooperative spirit. No HSK inspectors are resident at the plants. There are designated liaison personnel for each plant within HSK, who make periodic checks of plant conditions and procedures. Unannounced inpections are permitted, but are unusual, as both parties prefer that the technical agenda for each inspection be established in advance.

As utilities are responsible for the technical state of the plant, many inspections are conducted on their own initiative. Each plant is primarily responsible for its own safety; HSK oversees the utilities and ensures that these responsibilities are carried out. If HSK were dissatisfied with the quality or extent of a plant's particular testing or inspection procedure, the regulator would require that the routine be repeated. In sensitive cases, such tests might be supervised or assisted by an HSK (or an HSK-appointed) expert. In general, however, HSK does not keep abreast of the many detailed questions requiring attention in each plant. Thus, it is the utilities that must be the first to recognize and address technical problems.

Some specific examples of regulator/utility interactions may be given. At the Leibstadt plant for instance, batteries for powering emergency equipment were found to be defective; replacements were ordered.

Because these batteries were required to withstand earthquakes of a certain intensity, the replacements would not be easy to acquire. In fact, it would be one month before the equipment qualified for the task would arrive. Because the plant could not legally operate without such back-up power, the plant staff asked HSK for permission to operate using ordinary batteries (not earthquake-proven) for the 30 days until the new ones arrived. HSK allowed this modification of standard procedures.

At the Leibstadt and Gösgen plants, a special emergency heat removal system (SEHR) had been installed, separate from the standard low pressure emergency core cooling system (ECCS). For these plants, permission was requested to perform maintenance on the ECCS. The work would require the temporary disabling of the ECCS during the operation of the plant. Ordinarily, such an action would be imprudent. Because of the redundancy, however, the request was granted. Some minor amendments to the technical specifications were required. These examples illustrate HSK's general flexibility in rulemaking and receptivity to utility requests.

Conversely, utilities appear to have an open ear for regulatory proposals. In the case of the SEHR system at Leibstadt, this system was actually first proposed by HSK to the utilities. Most agreed with HSK that such a capability was a sound idea and should be installed in their plants. As demonstrated above, such extra requirements are often accompanied by compensating regulatory flexibility.

A case of a disagreement between HSK and a utility concerned design problems with the Leibstadt plant. This impasse ultimately resulted in a considerable construction delay. In the utility's opinion, certain regulatory rules were, at first, not sufficiently explicit regarding standards of design and review. The utility also questioned the document control procedures observed by HSK prior to construction.

Once construction was underway, HSK's safety requirements became more explicit and demanding; new standards led to some redesign. The design basis established in a 1973 contract between Leibstadt and the supplier was no longer adequate. The utility also objected that some equipment orders for the plant had been significantly delayed by HSK design reviews. In the utility's perspective, HSK was motivated to become more demanding by external political pressures, including some from neighboring nations. HSK learned and discovered more detailed information about the plant's design and construction. The regulator took issue with many of the newly revealed design characteristics, thus causing the delay. The utility nonetheless attributed the delay to political forces rather than viewing it as a direct result of HSK's actions.

HSK's views on the Leibstadt construction delay are somewhat different. According to the regulators, the plant incorporated a new containment design which had not been used before. In addition, the design of the nuclear island was new, both to the supplier and to the regulator. The utility began plant construction without awareness of the potential problems that these new designs could present. Regulatory officials objected that the utility did not allow them sufficient time to check the proposed design before proceeding with construction. As a result, HSK's standards were still evolving while the plant was being built.

In addition to its relationship with utilities, HSK is affected by governmental actions and, to a more limited extent, by the public. HSK officials can have their actions scrutinized by a political proceeding. Hearings to gather information are fairly common. For these meetings, HSK staff may be required to testify or to supply written statements. Dismissals or other extreme or capricious actions, however, are very rare.

Members of Parliament (MPs) can request that HSK provide answers to specific questions on plant safety. Some HSK staff members usually travel to Bern to assist the Energy Minister in answering the questions of the Parliament. The government also controls HSK's budget. This power has not been used as a tool of anti-nuclear sentiment, as has happened occasionally in the United States. In general, MPs say privately to HSK that they favor more money for the regulatory body. As the regulatory budget has not been prominent on the public agenda, budget increases have not been formally discussed. An increase in personnel was recently requested by HSK, which it received.

The Swiss public has one main avenue by which it may require HSK action. This opportunity is intended especially for those people living within a certain radius of a nuclear plant (including, incidentally, those German citizens close to a Swiss plant.) These people may require answers of HSK to questions affecting their health and safety. The technical information contained in the permit application, accompanied by safety reports, is usually adequate. Hence, additional safety analysis motivated by public hearings has not caused delays in plant construction or start-up.

### 5.3 Outage Reporting and Classification

#### 5.3.1 General Principles and Examples

The Swiss industry reports no regulatory outages. Similar to the practice of France and Sweden, problems that arise, whether LCO violations or needed modifications, are classified as technical difficulties and never as regulatory outages.

An illustration of outage reporting may be cited. At Leibstadt, there was a scram due to operator error. This caused a short outage, which was attributed to human error. Upon start-up, a safety relief valve and discharge line were found leaking.

The leak was not an urgent safety issue; Leibstadt chose to defer the repair. This decision was acceptable to HSK. An inspection, however, was requested by HSK one week prior to the repair in order to identify the leak's precise location and to gauge the extent of the work required. Leibstadt complied with this wish, but would have performed the repair without a preliminary inspection. Neither the repair nor the additional inspection time, however, was considered a regulatory loss. Rather, they were viewed as technical issues for which the utility was responsible, and classified accordingly.

At the Leibstadt plant in 1986, only 0.17 EFPH was lost due to testing. The tests conducted were of the MSIV and the turbine inlet valve. At first, testing the turbine inlet valve required reducing the plant's power to 90%. Even with this relatively small loss, the utility was able to reduce the EFPH lost still further in later tests. This was accomplished by learning the plant's sensitivity to scrams, test by test. In subsequent turbine valve tests, the utility only needed to lower power to about 98%. For the year 1986, Leibstadt had only 1.66% capacity lost due to forced outages.

### 5.3.2 Technical Specifications/LCO Violations

Swiss technical specifications, especially those developed for newer plants, are very similar to those in the United States. Swiss LCOs were basically derived from US values and remain quite similar to them today. Older specifications, while also using US vendor information as a baseline, are tailored to the individual designs of the earlier Swiss plants. Important differences in the regulation of old and new plants arise from greater redundancy in the newer plants. This feature has lengthened the inspection interval and the allowable inoperability time for many components.

### 5.3.3 Inspections

Swiss surveillance requirements are also of similar stringency to the US rules. The chief difference is that the Swiss requirements do not incorporate the variable surveillance interval found in the US documents. Both plant operators and regulators see some countervailing considerations that limit both the scope and stringency of surveillance requirements. First, there is the economic incentive to keep capacity losses due to inspections reasonably low. Second, inspection time is minimized to prevent plant employees conducting the checks from receiving more of a radiation dose than is absolutely necessary. Last, components are assigned strict inspection priorities to prevent the diversion of technical attention and expertise from the most important systems. Since the employees' time is a scarce resource, it must first be concentrated on the weakest and most vulnerable systems.

HSK has very rarely asked for mid-cycle shutdowns for inspections. The few that have been ordered were to verify the adequacy of earlier repair work performed during the annual refueling outage. An example of such an outage occurred at the Mühleberg plant in January 1986. An inspection of the stainless steel recirculation piping was required. This piping was to be replaced in the refueling outage later that year;

thus, the utility was likely not inclined to check the piping on its own initiative, in the absence of signs of trouble. HSK allowed the work to be scheduled at the utility's convenience. Officially, this inspection was motivated by a temporary change in the technical specifications; here, this was tantamount to the implementation of additional safety precautions. The inspection revealed that the earlier repairs were intact; no further corrective measures were necessary prior to pipe replacement.

### 5.3.4 Modifications

When a new problem is discovered at a plant, the principle of utility responsibility for plant safety governs the responses of both HSK and the utilities. HSK has the legal right to shut the plant down, but such extreme action has never occurred. In responding to problems, the utility is not normally faced with strict time limits, unless the situation's urgency demands it. Rather, management takes the time to formulate a measured, careful response that they can stand behind in confidence. The utility's package of solution strategies, including its proposed modifications, is presented to HSK for discussion and approval.

Utilities that take the initiative in presenting appropriate solutions are in a better position to control the modifications required of their plants. By taking the lead in problem resolution, it is easier to resist the introduction of unproven experimental procedures and hardware or unnecessary complexity by a well-meaning but meddlesome regulator. This is not to say that HSK is meddlesome. On the contrary, much of the regulatory burden has been lifted from HSK by the utilities' technical competence and vigorous efforts toward trouble-free plants.

HSK's actions and policies are consistent and harmonious with the utilities' behavior. If a problem is brought to its attention (by an event at a nuclear plant elsewhere in the world, for example), HSK would ask each utility to render an opinion explaining how the foreign development is relevant to its plants. HSK and the utilities keep informed of international nuclear events and research. For example, both groups carefully followed the international problems with stress corrosion cracking and reactor trip systems.

The primary response to the TMI accident by HSK was the establishment of an independent internal study group to examine critically various TMI-related issues. In particular, the study group was to address the problem of hydrogen formation and the possibility of filtered containment venting for pressure relief. A group of expert specialists outside of HSK was also convened for consultation on these topics.

Remarkably, no outages were experienced in Switzerland as a result of TMI. There was, however, a delay in the initial start-up of one plant after the accident. No modifications or other delays related to TMI caused any outages after the start of commercial operation. HSK required some modifications, similar to the new US requirements, for operating plants which arose from the TMI accident. Again, these changes were effected during the annual refueling outage, causing no further shutdowns or deratings.

## 5.4 Comparisons with US Experience

When comparing the Swiss and US nuclear industries, one must keep in mind that the Swiss system is an order of magnitude smaller in terms of the number of plants. This difference in scale accounts for many disparities between the two nations. For example, it is much easier for the HSK to be consistent in its treatment of Swiss utilities than for the NRC to do likewise in the United States. The five administrative regions of the NRC may encourage the same close relationship between regulator and utility as exists in a smaller system. This division of the NRC, however, could lead to interregional inconsistencies which confound the uniform implementation of regulation on a national scale. Also because of the size of the US industry, it is infeasible for the NRC to oversee certain critical test procedures at plants. HSK is able to send support personnel or an expert consultant if requested for especially sensitive work.

Despite these differences in size, the Swiss and US nuclear industries share the characteristic of having a wide variety of plant designs. HSK is able to respond effectively to this diversity with functionally oriented regulations; NRC requirements, in contrast, are generally far more prescriptive.

Swiss LCOs were modeled after those in the United States; they remain substantially similar today. Like France and Sweden, Switzerland considers the observation of the LCOs merely good practice. Thus, violations are technical problems in Switzerland, not regulatory outages as in the United States.

Inspection requirements in Switzerland include no provision for a variable surveillance interval, resident inspectors, or IEBs. Furthermore, where the redundancy of safety systems exceeds that in the United States, longer times of inoperability are permitted in Switzerland. Perhaps as a result, very few mid-cycle inspections have ever been required.

Modifications in Switzerland (including those for TMI) were accomplished without incurring any forced outages. This contrasts sharply with the US experience of a substantial number of forced outages for modifications, especially in the wake of TMI.

## 6.0 Conclusion

### 6.1 Data Reclassification and Conclusions

In the last three chapters, significant differences in European and US outage classification practices were revealed and discussed. Now, in the final chapter, the central question of this investigation will be put to the test. Knowing more about European outage classification conventions, the US data will be reclassified according to European standards. This adjustment will yield a more accurate picture of the burden of US regulation relative to that in the other nations studied.

As expected, because of highly similar technologies, the plant systems most troublesome in the United States were also those causing the most outages in Europe. More importantly, the same "regulatory" constructs and practices observed in the United States (e.g., technical specifications, surveillance requirements, and required modifications) were in place in Europe. Furthermore, the stringency of the US and European requirements is comparable. The difference, however, between the US and European systems is fourfold:

1) In the European nations studied, the voluntary operating standards and practices of the utilities

are apparently at least as stringent as safety regulations in those nations. Thus, a shutdown may be required when a plant exceeds these operating limits established by industrial practice, before those limits imposed by external regulation are reached.

2) The European nations consider capacity lost due to operating limits as a forced outage, rather than as a regulatory outage, as in the United States.

3) In the US, safety regulation imposes losses on utilities, through forced outages in particular, in a manner in which European regulation does not.

4) Frequent discussions and interactions occur between utilities and regulators in Europe. This dialog takes place between technical people and is characterized by professionalism and respect. The regulator/utility relationship has none of the litigious or antagonistic atmosphere seen in the United States.

The third point above highlights a key distinction that must be made here between regulatory behavior, on the one hand, and outage classification on the other. Regarding scheduled outages, the regulators of all four nations in the study have issued orders resulting in scheduled outages. Only in the United

States (and Sweden, to a much lesser extent), however, have these outages actually been classified as regulatory.* Thus, largely similar regulatory behavior on both sides of the Atlantic** has met with distinct classification methodologies for resulting outages. In reclassifying the US regulatory outage data, scheduled outages will be subtracted from the regulatory loss totals to make the data conform more closely to European conventions.

For forced outages, the picture is different. This investigation found no evidence that the European utilities studied had ever been faced with a regulatory order requiring a forced outage. This may indeed be the reason why no forced regulatory outages are reported in France or Switzerland, and only a few in

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*Sweden's regulatory losses, as noted in the fourth chapter, come as a result of technical differences of opinion between the regulator and a utility. The Swedish scheduled losses are not classified as regulatory quite as readily and arbitrarily as are some US scheduled losses. In Sweden, a bona fide disagreement must exist; in the United States, any regulatory order (an IEB, for example) regardless of utility opinion, may cause a regulatory outage.

**Regulatory behavior among nations is not completely comparable for scheduled outages, although less international variation is observed than for forced outages. The NRC often stipulates a shorter time horizon for outage planning than do Europe's regulators. Refer to the discussion following Tables 6.1 and 6.2 for further information.

Sweden.* Again, in contrast, the United States reports a substantial amount of regulatory forced outages. Here, the transatlantic differences in outage classification may be indicative of true disparities in regulatory behavior. The data do not show conclusively how the European utilities would respond to a non-negotiable demand for a forced outage. In particular, it is unclear how any resulting outage would be classified.

Because of these uncertainties, it will be assumed that the US forced outages result from regulatory actions that are not observed in the three European nations. Thus, in recasting the US data, forced outages alone will be treated as unique and bona fide sources of regulatory loss. In general, the reclassification will observe this rule. Namely, only the regulatory loss unique to the United States (in particular, the forced outage component of regulatory loss) will be included in the reclassification.

A different rule will be used for LCO violations and outages due to the unavailability of safety-related equipment. For these, neither scheduled nor forced

*Concerted efforts are made by the regulator and utilities in Sweden to schedule needed repairs at a time that is best for the utility. This policy is apparently successful in achieving far more scheduled than forced regulatory outages. Many of the extensive discussions between SKI and the utilities would be pre-empted by forced outages.

outages will be included in the reclassification. This is because the three European nations, without exception, treat safety readiness and the observance of LCOs as an integral part of industry practice, rather than as burdensome impositions by the regulator.

The reclassified US data appear in Table 6.1 (for PWRs, excluding TMI) and Table 6.2 (for BWRs) on pages 96 and 97, respectively. These include the original data presented in Tables 2.1 and 2.2, respectively, for comparison. The downward revisions discussed above are reflected in the reclassified data. Note that the scheduled outage component is also subtracted from the lesser outage categories -- licensing, fuel and core safety restrictions, BWR fuel limits. These causes of regulatory loss were not addressed in this study due to their small effect. It is assumed, however, that as with the major outage causes, only the forced outages are unique to the United States. Therefore, only this component is included.

From Tables 6.1 and 6.2, it can be concluded that international differences in safety regulation are not the primary source of the performance difference between the United States and the European countries studied. The recalculated totals for US regulatory loss are comparable to losses in other countries studied, from Table 1.1. The data represent an

TABLE 6.1: Reclassified Average Annual US Regulatory Capacity Loss, 1975-1984, for PWRs, excluding TMI (in percent)
Classification by regulatory outage cause

| Outage Cause | Capacity Loss (%) (as originally stated) | Capacity Loss (%) (as reclassified) |
|---|---|---|
| NRC-originated inspections | 3.47 | 1.15 |
| NRC-originated modifications | 2.33 | 0.63 |
| LCO violations | 1.86 | 0.0 |
| NRC licensing proceedings & hearings | 0.61 | 0.24 |
| Fuel and core safety restrictions | 0.04 | 0.02 |
| Combination | 0.04 | 0.02 |
| Unavailability of safety-related equipment | 0.03 | 0.0 |
| TOTAL | 8.38% | 2.06% |

(Combination category comprises inspections or LCO violations or modifications in combination with a non-regulatory cause.)

Source: OPEC-2 Database

TABLE 6.2: Reclassified Average Annual US Regulatory Capacity Loss, 1975-1984, for BWRs (in percent) Classification by regulatory outage cause

| Outage Cause | Capacity Loss (%) (as originally stated) | Capacity Loss (%) (as reclassified) |
|---|---|---|
| NRC-originated modifications | 4.45 | 0.41 |
| Combination | 2.86 | 0.25 |
| NRC-originated inspections | 1.61 | 0.53 |
| LCO violations | 0.86 | 0.0 |
| Fuel and core safety restrictions | 0.32 | 0.31 |
| NRC licensing proceedings & hearings | 0.16 | 0.03 |
| BWR fuel limits, i.e., MCPR, MAPLHGR | 0.08 | 0.05 |
| Unavailability of safety-related equipment | 0.02 | 0.0 |
| TOTAL | 10.36% | 1.58% |

(Combination category comprises inspections or LCO violations or modifications in combination with a non-regulatory cause.)

Source: OPEC-2 Database

estimate of US regulatory losses, and should be interpreted in light of the two caveats discussed below.

By excluding all scheduled losses in the reclassified data, the amount of bona fide US regulatory loss is understated. Some of the scheduled losses arise from impositions by the NRC that would not occur in the other nations studied. These impositions are not as urgent as those eliciting forced outages; still, US utilities are not given the freedom to schedule their work optimally in these cases. The US data available in the OPEC-2 database do not usually discriminate between short-term scheduling (e.g., ten days ahead of time) and long-term scheduling (e.g., during the annual refueling outage, perhaps ten months away). Without such a distinction in the US data, it is not possible to more closely observe the European outage classification conventions.

A second, and countervailing qualification for the reclassified US data should also be remembered. The retention of all forced outages in Tables 6.1 and 6.2 results in an overstatement of US regulatory losses. It is improbable that the European utilities would respond with a regulatory classification for all forced outages. Their propensity to take technical responsibility for the plant would likely lead them to believe that some of the forced outages are recommended by good engineering practice.

No further conclusions may be drawn about the source of the performance discrepancy between Europe and the United States. Some generalizations about European utilities' attitudes and behavior may be made, however. European utilities are clearly responsible for:

- Maintaining a competent in-house technical capability

- Proper systematic monitoring and management of plant operation to anticipate and prevent problems

- Taking the initiative in proposing plans of action to the regulators as problems do arise

- Encouraging a detailed understanding of the operation and characteristics of individual plants

- Advancing the development and adoption of improved safety systems at their plants.

The cornerstone of the European approach to nuclear safety is that plant safety is first and foremost the responsibility of each utility. This tenet explains much of the Europeans' regulatory policy and industry practice.

Accordingly, European utilities exercise leadership and demonstrate a high level of technical competence in anticipating, recognizing, and preventing technical problems. Despite the utilities' exacting standards, difficulties with the plant do surface occasionally, in Europe as in the United States. When such problems arise, European utilities take vigorous initiatives in formulating solutions, cooperating with the safety regulators in a collaborative, constructive, and technical dialog.

As with US plant performance levels, the practices of US utilities vary widely. From this investigation, however, it is apparent that many US utilities do not embrace the same practices, policies, and attitudes as do European utilities. At the same time, one must recognize the structural disincentives existing in the United States which discourage more positive behavior. Economic, regulatory, and political forces external to the utility all contribute negatively to the environment of the nuclear industry. Nonetheless, it is clear from this study that US utilities could benefit themselves, the nuclear industry, and the public by judiciously assimilating some elements of the style of their European counterparts.

## 6.2 Further Work

This investigation demonstrated that the plant performance differential between Europe and the United States arises from sources other than safety regulation. As indicated by some of the descriptive passages in this report, a logical next step would be an international comparison of utility policies and practices. Differences here should be probed to determine if they explain the observed performance discrepancy. At best, an in-depth investigation of many aspects of utility management might attempt to correlate good performance with certain management styles and policies.

As a second layer of complexity, further studies could bring other organizations and influences into consideration. For example, attention could be focused on the effects of the legal environment in which the US industry operates, and the influence of organized public opposition in the form of intervenor groups.

Also of interest are the complex influences of economic regulation on performance. Here, specific topics worth examining include economic disincentives to good performance, the effects of capital and operating budget restrictions on plant quality and performance, and the sometimes contradictory objectives of economic and safety regulation.

## 7.0 References


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[3] Seth David Hulkower, The Effects of Regulation on the Performance of Nuclear Power in the United States and The Federal Republic of Germany (Cambridge, MA: MIT-EL 86-008, 1986), p. 75.

[4] Ibid., p. 2.

[5] Wilson, p. 18.

[6] Ibid., p. 37.

[7] Hulkower, pp. 17-18.

[8] E. Beckjord, et.al.; International Comparison of LWR Performance (Cambridge, MA: MIT-EL 87-004, 1987); p. 3-29.

[9] Comissariat à l'Énergie Atomique (CEA); The Protection and Nuclear Safety Institute, IPSN (Fontenay-aux-Roses, France: CEA); p. 18.

[10] Ibid., p. 2.

[11] Ibid., p. 3.

[12] Ibid., pp. 4-5.

[13] Swedish Nuclear Power Inspectorate (SKI), A Presentation of Our Activities (Stockholm: SKI), p. 15.

[14] Ibid., pp. 4-5.

[15] Ibid., p. 3.

[16] Ibid.

[17] Wilson, pp. 293, 295.

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Revision 10
Date - 4/84

## 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

1 - Miscellaneous 10
- 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+PH-1973-1974)
- 3. LPRM vibrations (BWR-4 1975-1976)
- 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) 4
- 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 0371 - 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 1293)
- 6. Control rods (BWR-1973-1974 inverted CR; cracking)
- 7. MG sets/logic/relays/power supply/sequence controller 4
- 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 0739) | | 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 (B&W 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, BAT 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 1674) | | |
| | 9 - Components | | |
| | | 2. Shell/casing | |
| | | 3. Air ejector | |
| | | 4. Waterbox (e.g., fouling see also 1674) | |
| | | 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-0359) (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-0346)
7. Main steam bypass (to condenser) + BWR dump to condenser hotwell 5



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 (B&W - BWST; BWR - DWST see 0538) | 7 |
| | | 8. BIT | 1 |
| | | 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-1925)(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) | 2 |
| | | 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 2237) 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)
3. Auxiliary bldg. (e.g., Salem 1980) 9
4. Main steam tunnel (see also 1285 or 1373) 9

8 - Electrical 4
2. Cable routing
3. Cable splices and electrical connectors 9
4. Browns Ferry fire 10
5. San Onofre 1 cable fire 10

25 Economic 10

7 - Fuel economic 10
2. Coast to refueling/fuel depletion 10
3. Fuel conservation

8 - Grid economic
2. Low-system demand/spinning reserve 10
3. Load following 10

99 Left Over

## FOURTH LEVEL: PROBLEM MODE/EXTENT

00 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 4
- 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 3

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)
- 95 Structural/design/construction inadequacy 5

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

**Leakage/Crack**

04 External leakage (packing/seals/gaskets)
14 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 — these include procedural inadequacies | |
| 50 | 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) | 8 |

**BWR Fuel Limits**

| | | |
|---|---|---|
| 30 | CR seq. or pattern change ( every 5 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 | Combination: | 62 (drive) & other 5th-level coding | 10 |
| 92 | " | 07, 08, 09, 50 (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)
1 Manual (includes manual shutdowns with a unit trip at low power 2
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
6 Continuation of other outage (no interstitial electrical generation)* 9
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
2 SI signal generated (only one per SI generated--with most immediate cause)
3 There are related transients or significant subsequent events following this event (must always precede event with 8 in 6th level)
4 2 + 5 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)*
47 Other 4

**Electrical Power Transients**
40 Loss of power on an auxiliary bus (6.9 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)*
47 Other 4

**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*

EIGHTH LEVEL: SIGNIFICANT TRANSIENT INITATOR, (Cont.)

Feedwater, Condensate, Circulating Water and CVC System Transient, (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)

Reactor Coolant Pump/Recirculation Pump Transients

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

Reactor Coolant System Pressure and Temperature Transients or SI or RXT

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 flucuations 6
60 Pressure and/or level fluctuations including bubble in RCS (no trip necessary) 9
65 Inadvertent SI or spurious SI signals (HPCI pump start - BWT)*
66 Accidental depressurization of the main steam system (e.g., stuck open RV) (see #34) (no trip necessary) 5
02 Reactor trip (no other initiating events noted) + spurious reactor trips 2

Control Rod Transients

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

Valve Malfunction Transients

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)

Reactor Coolant System/Steam Leak (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 051)
74 RCP seal failure
75 RCS system boundary valve failure* (leak past valve - see #72) (9L #062) 5
76 CRD mechanism housing rupture or CR ejection
77 Steam line break* (all sizes) (see 07,31)
78 FW pipe break* (all sizes)

EIGHTH LEVEL: SIGNIFICANT TRANSIENT INITATOR, (Cont.)

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

Other Transient Initiators

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 - 104) (no trip necessary)
94 Loss of Service Water System (9th Level - 081)
95 Loss of Instrument Air System (9th Level - 101)
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 | |
| 052 | 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 |
| 039 | 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) | |
| 052 | Snubber or piping support out of compliance of failure | |
| 053 | MSIV failure (including leaks) | 8 |
| 057 | BWR MSIV leakage control system (valves & controls) | 7 |
| 054 | FW isolation valve failure | |
| 055 | CB vacuum pump failure | 2 |
| 264 | Containment Depressurization Actuation I+C failure (PWR) | |
| 507 | ADS actuation problems (BWR) | |
| 056 | CB Inerting System component failure (include gas/H2 analyzer) | 2 |

NOTE: For CB Spray see 530s for BWRs and 260s for PWRs 5

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) (BV - 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 S-95)
102 Fire Protection System component failure
103 Failed fuel detector failure
104 CCR habitability systems component failure (Level S-93)
105 Radiation Monitoring System component failure 2
106 Filter Exhaust System outside of CB - HVAC (BWR area coolers) 4
045 SBGTS (BWRs)

**Other**
120 Turbine stop valve (TSV) or CV does not close or slow response 4

## 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 3
214 RCS leak detection system 9

**LPSI Failures**
221 Accumulator failure/incapacitation
220 RHR pump failure 2
222 LP pump failure
223 System valve failure
224 I&C failure
225 Other failure

**HPIS Failures**
230 Charging pump failure 2
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 1
238 Upper head injection system valves
239 Upper head injection accumulator and other 9

**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 4
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 5
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 4
323 RHR I&C failure
324 Other RHR failure

ESS (EMERGENCY SAFEGUARDS SYSTEMS), Cont.

**BWR**

**ECI (Emergency Coolant Injection)**

411 System failure not listed below 9
412 Condensate Storage Tank (CST) failure/incapacitation
413 MUD-DSWT
414 Leak detection system component problem 7

**LPCIS + RHR Failures (LPCI/RHR/VCCS)**

421 Pump failure
422 System valve failure
423 System I&C failure
424 Other failure
425 RHR heat exchanger problem 7

**CSIS Failures (CS)**

431 (LP) core spray pump failure
432 System valve failure
433 System I&C failure
434 Other failure

**HPCIS Failures**

441 HP core injection pump failure
442 System valve failure
443 System I&C failure
444 Other failure

**Standby Liquid Poison Injection System**

451 SLPI system failure
452 SLPI tank failure
453 SLPI pump failure
454 Explosive plug valve failure
455 System I&C failure
456 Other failure

**RCIC Failures + Isolation Condenser System Failures** 5

461 RCIC pump failure
462 Isolation condenser failure
463 System valve failure
464 System I&C failure
465 Other failure

**VS (Vapor Suppression) System**

481 System failure
482 Torus support cracks
483 Other torus problem
484 Torus DP problem
485 Vacuum breaker problem (drywell to torus) 5
486 Other VS problem
487 Vacuum relief problem (CB to RxB) include permissive problems 7

**FW (Feedwater System**

491 System failure
492 Feedwater pump failure
493 Condensate pump failure
494 System valve failure
495 System I&C failure
496 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 4
507 ADS actuation problems 7

**Other**

520 Component failure in dump to condenser Hotwell System 5

**Containment Spray System (BWR-2s)**

530 CSS System failure
531 CSS pump failure
532 CSS valve failure
533 CSS heat exchanger failure 5
534 CSS I+C failure
535 Other CSS component failure

## 8.2 Glossary of Abbreviations Used

| | |
|---|---|
| ACRS | Advisory Committee on Reactor Safeguards |
| BWR | Boiling Water Reactor |
| CEA | French Atomic Energy Commission (Commissariat à l'Énergie Atomique) |
| CF | Capacity Factor |
| EAF | Energy Availability Factor |
| ECCS | Emergency Core Cooling System |
| EdF | Électricité de France |
| EFPH | Equivalent Full Power Hours |
| FF | French Francs |
| FRG | Federal Republic of Germany |
| HSK | Swiss Federal Nuclear Safety Inspectorate (Hauptabteilung für die Sicherheit der Kernanlagen) |
| IAEA | International Atomic Energy Agency |
| IEB | Inspection/Enforcement Bulletin (US) |
| IPSN | Protection and Nuclear Safety Institute (Institut de Protection et de Sureté Nucléaire) |
| KSU | KärnkraftSäkerhet och Utbildning (Nuclear Training and Safety Center) |
| LCO | Limiting Condition of Operation |
| LWR | Light Water Reactor |
| MAPLHGR | Maximum Average Planar Linear Heat Generation Rate |
| MCPR | Minimum Critical Power Ratio |
| MIT-EL | Massachusetts Institute of Technology - Energy Laboratory |
| MP | Member of Parliament |
| MSIV | Main Steam Isolation Valve |
| MW | Megawatts |

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|---|---|
| MWe | Megawatts (electric) |
| NRC | Nuclear Regulatory Commission (US) |
| OECD | Organisation for Economic Cooperation and Development |
| OPEC-2 | Operating Plant Evaluation Code - 2 |
| PWR | Pressurized Water Reactor |
| RCC | Design and Construction Rules (France) |
| RKS | Nuclear Safety Board of the Swedish Utilities (Rådet för KärnkraftSäkerhet) |
| SCSIN | Central Service for the Safety of Nuclear Installations (Service Central de Sureté des Installations Nucléaires) |
| SEK | Swedish Kronor |
| SF | Swiss Francs |
| SKI | Swedish Nuclear Power Inspectorate (Statens Kärnkraftinspektion) |
| STB | Surveillance Test Book (Sweden) |
| TMI | Three Mile Island |
| US | United States |