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IDENTIFICATION OF IMPORTANT "PIUS" DESIGN CONSIDERATIONS AND ACCIDENT SEQUENCES USING QUALITATIVE PLANT ASSESSMENT TECHNIQUES*

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ABSTRACT

The PIUS (Process Inherent Ultimate Safety) reactor is an advanced design nuclear power plant that uses passive safety features and basic physical processes to address safety concerns. Brookhaven National Laboratory (BNL) performed a detailed study of the PIUS design for the NRC using primarily qualitative engineering analysis techniques. Some quantitative methods were also employed. There are three key initial areas of analysis: FMECA, HAZOP, and deterministic analyses, which are described herein.

Once these three analysis methods were completed, the important findings from each of the methods were assembled into the PIUS Interim Table (PIT). This table thus contains a first cut sort of the important design considerations and features of the PIUS reactor. The table also identifies some potential initiating events and systems used for mitigating these initiators.

The next stage of the analysis was the construction of event trees for each of the identified initiators. The most significant sequences were then determined qualitatively, using some quantitative input. Finally, overall insights on the PIUS design developed from the PIT and from the event tree analysis were developed and presented.

1. INTRODUCTION

The PIUS reactor was designed by ABB Atom over the past decade. ABB has requested an NRC licensability review in accordance with the Policy Statement on Advanced Reactors set forth in NUREG-1226[1]. As part of this request ABB submitted a Preliminary Safety Information Document (PSID)[2] to provide the basic information necessary for a preliminary review of the conceptual design of PIUS. This preliminary review is sometimes referred to as a preapplication review and would likely be followed by a detailed Safety Analysis Report (SAR) written by ABB and submitted in turn to the NRC as part of a design certification request in accordance with 10CFR52[3]. The results presented here are based on the design as given in Reference 2. It is recognized that the design continues to evolve and that changes are currently being contemplated. which would address some of the concerns noted herein.

In support of the U.S. Nuclear Regulatory Commission preapplication review of the Process Inherent Ultimate Safety reactor (PIUS 600), BNL is providing an integrated analysis of the reactor design. The objectives of the analysis are to:

* This work done under the auspices of the U.S. Nuclear Regulatory Commission.

- perform systems research on the PIUS reactor design,
- aid the NRC staff in understanding of the dynamic operation,
- identify any potential safety problem areas that need to be addressed during the plant certification review,
- follow vendor and independent research projects related to the PIUS design,
- recommend research and engineering areas that may be needed, and
- remain cognizant of PIUS design and safety developments.

This paper presents the methodology used for the study and, summarizes the findings of the PIUS analysis, which was documented in a BNL report entitled, "Integrated Systems Analysis of the PIUS Reactor"[4].

2. PLANT DESCRIPTION

PIUS is a pressurized water reactor (PWR) with a nominal power rating of 2000 MWt. The primary system consists of four loops, each with a main circulating pump and steam generator. The four loops are connected to a flow structure that cools and moderates the reactor core. This flow structure, including the core, is submerged in a large pool of cool, heavily borated water that, in an automatic and passive manner, provides both negative reactivity for shutdown and cooling water for decay heat removal. Other significant characteristics of the PIUS design are:

- A prestressed concrete reactor pressure vessel (PCRV) containing 3300 cubic meters (800,000 gallons) of borated water in a pool outside of the flow structure,
- No control or shutdown rods,
- Normal reactivity control (and power control) is provided by boration of the primary loops,
- The active (or automatic) scram trips a main circulating pump resulting in an imbalance in the flow rate through the steam generator loop and the flow rate through the core produced by heating the coolant. When this occurs, the pool loop is activated. The highly borated water shuts down the core and continued flow through the pool loop removes decay heat,
- A manual scram also trips the main circulating pump with similar results,
- A passive scram results, when any transient imbalances the flow through the primary loops and the core, causing inflow of the highly borated pool water into the reactor,
- Flow from the pool is prevented during normal operations by an upper and a lower density lock, which are activated upon a scram.

- The heat capacity of the pool is such that seven days of boiling from decay heat are required before the core will uncover. Water may be resupplied through connections external to the containment.

At 640 MWe PIUS has about half the power of current large conventional PWRs. The core height and the average linear heat rate is about two-thirds of a PWR. PIUS operates at a slightly lower temperature (54°F temperature rise through the core), and at about half the operating pressure of a typical PWR. Thus, in comparison with a conventional light water reactor, the design parameters would appear to put less stress on the PIUS reactor.

The reactor is started by blocking the pool loop flow with a gas bubble in a gas cap located above the lower density lock. Under computer control, the water in the primary loop is deborated. As power in the core increases, core flow increases. The primary flow is also increased to maintain the flow balance. When the reactor reaches 20% power, the gas bubble is removed and the flow balance is maintained by computer control of the main circulating pump (MCP) speed. This maintains the interface between hot primary water and the colder pool water in the lower density lock. Deboration continues, as the MCPs increase speed, until full power is reached.

A controlled shutdown is accomplished by the addition of boron, while the computerized servo-control reduces pump speed to maintain the flow balance. When the MCPs reach 20% of full speed, they are stopped, activating the borated pool loop to complete the shutdown of the reactor. While in a hot or cold shutdown situation, PIUS is passively cooled by natural circulation flow to the core through the upper and lower density locks from the reactor pool. Active cooling of the pool is provided by pumping the water through heat exchangers to the heat sink. Passive pool cooling is also available via closed-loop natural circulation heat-exchangers to heat sinks located on the reactor building roof.

3. METHODOLOGY

An overview of the methodology used in the BNL PIUS study is shown in Figure 1. Briefly, the initial part of the study consisted of analyzing the PIUS design using three very different methods: FMECA, HAZOP, and Deterministic Analyses. Each of these will be discussed, in turn, below.

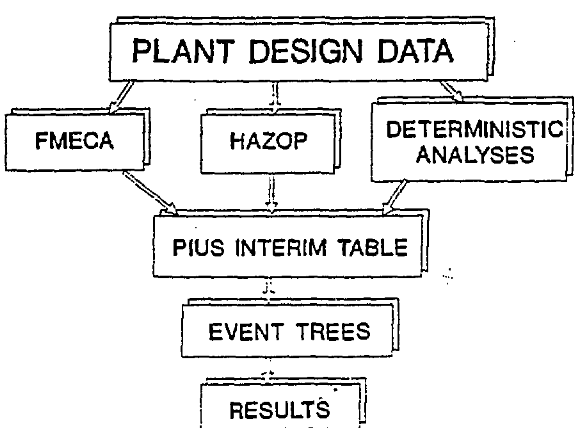

Figure 1

3.1 FMECA

FMECA or Failure Modes, Effects and Criticality Analysis is an inductive technique applied to systems, sub-systems, and components to identify safety-significant failures.

Each component of the design is identified and its purpose is explained. A failure mode is identified along with the causes of failure. The effect of this failure on the system and plant is assessed, followed by a qualitative risk estimate called criticality, and a qualitative frequency estimate for this failure mode. This frequency estimate may be based on actuarial data, part count, or reliability class. Mitigative factors are stated, followed by observations that are made in performing the analysis. This process is repeated until all parts of the plant, deemed to be important for public and worker safety, have been analyzed. The results are presented in a tabular format.

The FMECA table consists of a major title followed by a line that identifies the system and the subsystem being investigated (See Figure 2) The table has nine columns. The left column contains numbers that identify each of the analyses. In some cases, a number and a letter are used to identify analyses in which different failure modes are analyzed, e.g., a valve failing open and the valve failing closed. This column is followed by a description of the item and its function. The third column contains the postulated failure mode, e.g., fail open. The fourth column lists the causes for the failure mode, e.g., binding, distortion, corrosion or power failure. This column may contain multiple failures that may be necessary for the failure of this component. The fifth column explains the effect on the system and plant. Criticality, listed in column six, is a qualitative risk estimate, a combination of consequences and frequency of occurrence. The criticality categories used in this work are:

- High - further consideration is recommended
- *Medium - possible significant*
- Low - not believed to be risk significant.

Constructing a FMECA

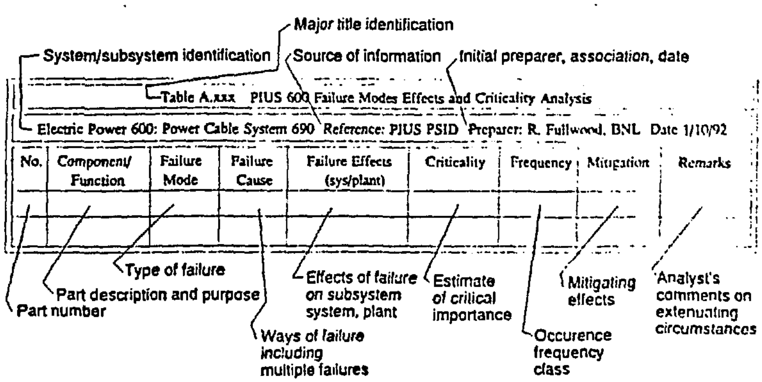

Figure 2 - Tabular Form of the FMECA

The estimated frequency of occurrence of a failure used in column 7 is based on the categories shown in Table 1, which were given in the PIUS PSID.

Table 1 - Frequency Categories

FREQUENCY CATEGORIES:

These categories are used for both failures or occurrences per year and for failures per demand.

I	Frequent: 10 to 1
II	Anticipated: 1 to 0.1
III	Rare: 0.1 to 0.01
IV	Very rate: 0.01 to 1E-4
V	Extremely Rare: 1E-4 to 1E-7
VI	Incredible: Less than 1E-7

The mitigation factors are presented in the next column. These factors could include equipment providing redundancy, human actions, or considerations mitigating the effects of the failure. The last column gives remarks made by the analyst regarding assumed conditions for the analysis, or suggestions for improving the design.

3.2 HAZOP Analysis

Concurrent with the FMECA, deductive Hazards and Operability (HAZOP) studies were performed. HAZOP[5] is an accident detection and prevention technique used by the Chemical Process Industry. BNL used HAZOP for analyzing PIUS-600 to provide a deductive method, which is different from, and complementary to the FMECA. Using both methods leads to more complete accident identification than either alone. In using the HAZOP technique, the team first selects a plant operating mode (e.g., startup) and then a process node (e.g., the core inlet) within the plant systems for detailed analysis. Next, a parameter is selected for analysis at the process node (e.g., core inlet temperature or flow rate). Perturbations are postulated in the parameter (e.g., low flow or high flow) and possible causes for the perturbations are examined. Finally, the identified causes are analyzed to determine ramifications of such an occurrence beyond just the initially postulated parameter perturbation. One can see that a full application of HAZOP requires detailed plant operating and performance information. Thus, a full and complete HAZOP was not possible for PIUS at this stage of design development. Nevertheless, the application of HAZOP in this situation was useful and informative.

3.3 Deterministic Analyses

Both the FMECA and the HAZOP were supported and complemented by deterministic engineering analyses such as: thermal hydraulics, seismic and stress analysis, and materials evaluations.

A PC-based thermal-hydraulic and reactor physics model of the PIUS primary systems was developed. This model, called the PIUS interactive plant analyzer (PIPA) includes the reactor core, the primary coolant piping, steam generators, main circulating pumps, pressurizer, density locks, the large borated pool, and the control and protection systems. PIPA was used for preliminary quantitative analysis of the reactor and primary systems during transient and accident conditions. Some events analyzed include: full scram, loss of heat sink, depressurization transient, and uncontrolled boron dilution.

Deterministic engineering analyses were performed to evaluate in preliminary fashion, the susceptibility of PIUS to various external events, such as fires, floods, and earthquakes. Some areas meriting further evaluation were identified. Additionally materials evaluations were performed to evaluate the potential of component degradation or failure during reactor operation. Factors considered in this evaluation included, high stress, cyclic stresses, elevated temperatures, neutron and hydrogen embrittlement, various types of corrosion, and mechanical creep.

3.4 Consolidation of Results

The preliminary findings from the three separate analysis techniques of FMECA, HAZOP, and deterministic engineering were consolidated into the PIUS Interim Table (PIT). The table provides an overview of the results and also was useful for entering the next phase of the study, namely the Event Tree Analysis. From the PIT, both accident initiating events and mitigating systems for these initiators were identified. This information was then supplemented by light water reac'or plant experience and generic probabilistic risk assessment (PRA) information to complete the list of initiators used in the event trees. Mitigators from the PIT were supplemented based on a knowledge of plant design from the PSID and were used to prepare event trees for each initiator.

A summary of the more notable preliminary findings to emerge from these three analysis techniques is presented here in Table 2. One should keep in mind a few points regarding these findings. First of all, the event tree analysis described below, reinforced some of the findings. Also, additional insights beyond this table were obtained from the event tree analysis. Importantly, one must also recognize that in some instances the information available for review in the PSID was not detailed enough, at the current stage of the PIUS design, to answer all concerns. Thus some issues may be raised here which will not be concerns when the more detailed information becomes available. Further, due to the evolving nature of the design, some areas identified in this table are already in the process of redesign, which should address and eliminate some of the concerns noted.

3.5 Event Tree Analysis

Table 3 provides a listing of the initiators analyzed, ordered by frequency in occurrences per year. The frequencies are somewhat conservative estimates and are given as categories or ranges of frequencies, which are defined in Table 1.

The initiators in Table 3 are self-explanatory and often considered in PRA analysis of nuclear power reactors. One exception is computer malfunction (CM) which is included because PIUS lacks hard-wired control and because of the FMECA-identified concern for electromagnetic interference (EMI) associated with the thyristor control of the main circulating pumps.

Table 2 - Summary of Preliminary Concerns & Issues for Further Review
1. Lack of diversity in the scram - there is only one way to scram PIUS, that is by activation of the pool loop. Note: Information provided informally to BNL subsequent to this review indicates this may be addressed by design changes.
2. Density lock blockage - a scram stops the heat removal through the steam generators and activates the pool loop. Core decay heat cannot be removed if pool flow is blocked. Either continuous sensing of the ability of the pool loop to function if called upon or the ability to restore flow through the steam generators would help to mitigate this condition.
3. Lack of redundancy in the active scram - scram is performed by removing power from a dedicated pump. No redundancy such as usually found in LWR scram systems is indicated.
4. Power distribution - in the absence of control rods, corrections to flatten the power distribution do not seem possible.
5. Boron stratification could result in low-borated water entering the core region with the resulting reactivity transient.

Table 2 - Summary of Preliminary Concerns & Issues for Further Review
6. Common-cause computer failure - PIUS is dependent on digital computer control and information processing. The thyristor variable speed control of the main cooling pumps produces electromagnetic interference which could fail the computer equipment.
7. Failure to maintain adequate boration in the reactor pool - the concern is that the pool boration system may deborate the pool. Protection against this could be provided by monitoring the boron in the pool.
8. Seismic collapse of the flow structure - through structure-flow motion coupling fluid drag forces may be severe enough to damage the flow structure sufficiently to prevent core cooling.
9. Benchmarking of thermal hydraulic codes against integral test results is needed.
10. Inconsistency in safety classification of systems mixing non-safety with safety systems.
11. During refueling, the in core instrumentation is removed; no explanation is provided with regard to the approach to criticality.
12. Loss of heat insulation may result in entrained material that could cause loop-flow blockage.
13. Foreign objects that may enter during refueling or maintenance or loose parts may be causes of flow blockage of pool loop flow.
14. Failure of the reactor support structure could drive the gas cap down on the lower density lock and may block coolant flow.
15. A number of seismic design issues were identified, as summarized in Section 6.4.
16. Materials concerns were identified in the following areas: stainless steel pool liner, gas cover lock, wet thermal insulation, reactor internals, and boric acid leakages.
17. Status of Control Room (CCR) - Discussions on PSID seem to indicate low importance to CCR operators. CCR is designated non-safety grade.
18. Tendon inspection - Tendons sometimes lose tension in service. If a sufficient number do, then catastrophic PCRV failure may be possible. However, tendon inspection appears to be difficult.
19. Gas-lock blockage - the concern is that under degraded core conditions, gas may accumulate in the cap and block loop flow.
20. Accessibility for refueling - refueling PIUS requires flow structure disassembly and grappling fuel from a 47 m (154 feet), most of which is in water. The boundary fuel elements appear to not be accessible on a straight line.

Table 3 Ordering of Initiators by Frequency of Occurrence

Frequency Category	Initiator
I	Turbine Trip
I	Loss of Off-Site Power
II	Feedwater Transients
III	Computer Malfunction
III	Fuel Element Drop
IV	Steamline Break
IV	Flow Blockage During Refueling
IV	Small Primary Pipe Break
V	Large Primary Pipe Break
V	Severe Seismic Occurrence

Table 4 lists the important mitigating functions or systems that were used in the event tree analysis. A brief explanation of these will be given to help provide an understanding of the response of the PIUS reactor to transient and accident initiators.

Table 4 - Systems Ordered by Importance to Safety
Passive Scram (PS)
Active Scram (AS)
Manual Scram (MS)
Emergency Water Makeup (EWM)
Containment (CNM)
Remove Blockage (RB)
Critical Heat Flux (CHF)
Passive Pool Cooling (PPC)
Active Pool Cooling (APC)
Steamline Isolation (ISO)
Feedwater Reduction
Reactivity Control

There are five ways in which PIUS can achieve subcriticality and this accounts for five of the mitigators. The speed with which they add negative reactivity and shutdown the reactor varies considerably.

1) Active or automatic scram (AS) means a scram initiated by the Reactor Scram Trip System when it detects monitored parameters outside their allowable ranges. When this occurs, the computer in the Reactor Scram Trip System causes power to be removed from one MCP. This results in a flow imbalance and pool loop activation.

2) Manual scram (MS) is a scram initiated by an operator pressing a scram button. The subsequent actions are the same as for the active scram.

3) Passive scram (PS) is a scram that results from any flow imbalance in the reactor causing pool loop activation.

4) Hot shutdown is effected by a feedwater reduction causing the reactor coolant temperature to rise, shutting down the reactor by the strong negative reactivity temperature coefficient. This must be accompanied by boration of the primary coolant so that the reactor remains shutdown after cooling.

5) The reactor can be shutdown by boration of the primary coolant, using the Reactivity Control System.

When the pool loop is activated to scram the reactor, decay heat from the reactor core is also transferred to the pool. Active pooling cooling (APC) refers to the pumped circulation of pool water through heat exchangers outside of the PCRV to remove decay heat from the pool.

Passive pooling cooling refers to the passive method for removing decay heat from the pool. whereby water circulates through submerged heat exchangers via natural circulation to natural draft cooling towers on the building roof.

Should these two systems fail there is sufficient water in the pool to remove decay heat for several days with no operator action. Emergency water makeup (EWM) to the pool is also possible via manual operator action from an external source if necessary.

The containment (CNM) surrounds the reactor, primary systems, borated pool and PCRV. It is a pressure suppression type and is the last line of defense against the release of radioactivity to the public.

Two other mitigators, Steamline Isolation (ISO) and Remove Blockage (RB) relate only to specific initiators, namely Steamline Break and Flow Blockage during refueling. Finally, Critical Heat Flux (CHF) was added to some event trees such as Large Primary Pipe Break, to reflect the likelihood that critical heat flux would be exceeded on a depressurization transient.

For each of the identified initiators event trees were prepared. The frequency for initiators and the probabilities for mitigator failure were grouped into the Frequency Categories of Table 1, based on generic data. An example event tree for the Large Primary Pipe Break is shown in Figure 3.

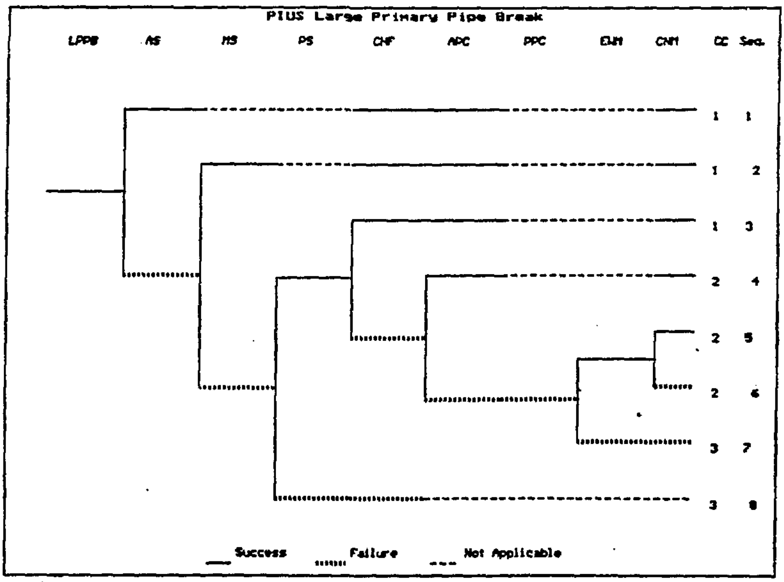

Figure 3 Event Tree for the Large Pipe Break Scenarios

The event trees were analyzed using the Brookhaven Event Tree Analyzer (BETA) computer program. This allowed sequence frequencies to be calculated, which were then placed into the frequency categories of Table 1. The results are shown in Table 5

Each of the sequences were analyzed to estimate the potential for core damage and release of radioactive materials. These results were qualitative using the consequence category definitions of Table 6.

Table 5 BETA Results for PIUS Large Primary Pipe Break (LPPB) Scenarios

Sequence Number	Sequence Probability Category	Consequence Category	NRC Event Condition
1	IV	1	3
2	IV	1	3
3	V	1	4
4	V	2	4
5	VI	2	4
6	VI	2	4
7	VI	3	4
8	VI	3	4
Total CC-1	IV	-	-
Total CC-2	V	-	-
Total CC-3	VI	-	-

The event trees and accident sequences were then reanalyzed, using the NRC Event Condition Methodology. This methodology classifies events by initiator frequency and combinations of failures. The event condition method was applied to the same event trees, thereby allowing the results from conventional analyses to be compared, with the event condition methodology. Event Condition (EC) 1 is termed an Abnormal operating Occurrence, EC2 is a Design Basis Accident, EC3 is termed Beyond Design Basis Accident, and EC4 are Residual Risk Sequences. Essentially as you proceed from EC1 to EC4 sequences you are dealing with lower frequency initiators and more failures of mitigating equipment. Table 5 also shows the classification of the 8 LPPB sequences using both the qualitative consequence categories and the NRC Event condition classification.

Table 6 Definitions

Consequence Categories:

CC-1 Conditions that result in exposures that do not exceed 10CFR20 limits.

CC-2 Accidents such as clad damage and releases inside containment resulting in exposures greater than 10CFR20 limits but less than 10CFR100 limits.

CC-3 Severe accidents (e.g., core melt) resulting in potential exposures greater than 10CFR100.

4. RESULTS

Results from the study are generally of two types, specific insights or findings and more general insights or prioritized lists from the event tree analysis. The key specific findings from the FMECA, HAZOP and deterministic supporting analyses are given in Table 2.

In order to summarize findings from the event tree analysis, sequences of various types were summed, importance calculations were performed, and results presented in ordered tables. Table 7 presents an ordering of sequence types (by initiator) according to their contribution to severe accident sequences (CC-3). Table 4 earlier provided a similar ordering of mitigating systems.

The most dominant sequences leading to CC-3 from Table 7 are: Loss of Off-Site Power followed by failure of the passive scram system, and Computer malfunction followed by failure of the passive scram system. This emphasizes the importance of the passive scram shown in Table 4.

Table 7 - Initiators Ordered by CC-3 Frequency	
Frequency Category	Initiator
IV	Loss of Off-Site Power (LOSP)
V	Computer Malfunction (CM)
V	Flow Blockage (FB)
VI	Feedwater Transient (FWTR)
VI	Small Primary Pipe Break (SPPB)
VI	Large Primary Pipe Break (LPPB)
VI	Steamline Break Inside of Containment (SBIC)
VI	Steamline Break Outside of Containment (SBOC)
VI	Turbine Trip (TTRIP)

Some conclusions of the study are that synergistic FMECA and HAZOP provide an in-depth analysis of the plant despite the lack of detailed information regarding the thermal-hydraulics under degraded conditions. Event trees provide a valuable extension to the FMECA/HAZOP/Supporting Analyses by showing how PIUS may respond to various accident sequences. The event trees have provided the bases for preliminary ranking of initiators and systems important to safety.

REFERENCES:

1. NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," June 1988.

2. PIUS Preliminary Safety Information Document (PSID), Volumes I, II, and III, dated July 1991.

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5. F. P. Lees, editor, "Loss Prevention in the Process Industries," Butterworths, London, 1980.