SIZEWELL B
NUCLEAR POWER STATION

Report to the Chief Inspector on the basis for the decision to grant consent for full commercial operation

December 1995
FOREWORD

This report describes the extensive work undertaken by staff of the Health and Safety Executive's HM Nuclear Installations Inspectorate in their assessment and regulation of the pressurised water reactor at Sizewell in Suffolk. It covers the period from licensing the construction of Sizewell B through to and including the granting of a consent to take the station into operation.

The task was the largest that the Inspectorate has ever undertaken and represents the culmination of over 100 staff years of effort in the period covered by this report. In accepting the report I wish to record my gratitude for the thorough and professional approach adopted by all staff involved in the project.

Sizewell B is now an operational nuclear power station. Like every other nuclear installation we shall continue our work to ensure that standards of safety are maintained and, where reasonably practicable, improved.

S A HARBISON
CHIEF INSPECTOR OF NUCLEAR INSTALLATIONS
# CONTENTS

<table>
<thead>
<tr>
<th>Section</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>INTRODUCTION</td>
<td>1</td>
</tr>
<tr>
<td>THE REGULATORY PROCESS</td>
<td>1</td>
</tr>
<tr>
<td>GENERAL WORKING ARRANGEMENTS</td>
<td>5</td>
</tr>
<tr>
<td>ASSESSMENT OF THE SAFETY CASE:</td>
<td>6</td>
</tr>
<tr>
<td>Introduction</td>
<td>6</td>
</tr>
<tr>
<td>Nuclear Safety Strategy</td>
<td>7</td>
</tr>
<tr>
<td>General Design Aspects:</td>
<td>8</td>
</tr>
<tr>
<td>Categorisation of Structures and Equipment</td>
<td>8</td>
</tr>
<tr>
<td>Design Limits and Loadings for ASME III Mechanical Equipment</td>
<td>10</td>
</tr>
<tr>
<td>Equipment Qualification</td>
<td>10</td>
</tr>
<tr>
<td>Hazard Protection</td>
<td>11</td>
</tr>
<tr>
<td>Quality Assurance</td>
<td>14</td>
</tr>
<tr>
<td>Achievement of Safe Shutdown</td>
<td>15</td>
</tr>
<tr>
<td>Human Factors</td>
<td>17</td>
</tr>
<tr>
<td>Reactor Core</td>
<td>19</td>
</tr>
<tr>
<td>Reactor Coolant System</td>
<td>21</td>
</tr>
<tr>
<td>Engineered Safety Features</td>
<td>27</td>
</tr>
<tr>
<td>Reactor Protection and Control Systems</td>
<td>29</td>
</tr>
<tr>
<td>Main and Essential Electrical Systems</td>
<td>36</td>
</tr>
<tr>
<td>Other Systems</td>
<td>38</td>
</tr>
<tr>
<td>Radioactive Waste Management</td>
<td>40</td>
</tr>
<tr>
<td>Radiological Protection</td>
<td>42</td>
</tr>
<tr>
<td>Fuel Storage and Handling</td>
<td>44</td>
</tr>
<tr>
<td>Civil Structures</td>
<td>46</td>
</tr>
<tr>
<td>Fault Analysis:</td>
<td>49</td>
</tr>
<tr>
<td>Probabilistic Safety Analysis</td>
<td>50</td>
</tr>
<tr>
<td>Transient Analysis</td>
<td>53</td>
</tr>
<tr>
<td>Decommissioning</td>
<td>56</td>
</tr>
<tr>
<td>Summary</td>
<td>57</td>
</tr>
</tbody>
</table>
Contents (cont'd)

REGULATORY INSPECTION ACTIVITIES:

Introduction
Regular (Basic) Inspections
Team Inspections:
  The Arrangements for Commissioning
  The Acceptability of Technical Specifications
  The Radioactive Waste Facility
  Readiness for Fuel Load
  Readiness for Operation
  The Operational Safety Case
  Completion of Commissioning
Specific (Reactive) Inspections
Monitoring Progress with Construction and Commissioning
Witnessing of Emergency Exercises
Joint Inspections
Exercising the Licensing Function
Other Activities
Summary

COMPLETION OF COMMISSIONING

PLANNED DEVELOPMENTS

CONCLUSIONS

REFERENCES

TABLES:

Table 1  Milestones in the Licensing Programme
Table 2  Licensing Stages in Construction and Commissioning
Table 3  NII Powers Under a Nuclear Site Licence
Table 4  Allocation of Resources (from 1987)
Table 5  Licensing Activity Summary Programme Topic Areas
Table 6  Hierarchy for Meetings between NII and the Licensee
Table 7  Standard Licence Conditions
Contents (cont'd)

FIGURES:

Figure 1  Sizewell B Power Station - Site Layout
Figure 2  Nuclear Fuel Assembly
Figure 3  Reactor Coolant System
Figure 4  Reactor Pressure Vessel and Internals
Figure 5  Reactor Coolant Flow Process
Figure 6  Steam Generator
Figure 7  Electricity Production Process
Figure 8  Pressuriser
Figure 9  Station Control and Protection Systems
Figure 10 Reactor Trip System - Voting Logic Arrangement
Figure 11 Main and Essential Electrical Systems
Figure 12 Main Steam System
Figure 13 Irradiated Fuel Handling Route
Figure 14 Reactor Building
INTRODUCTION

1. The Sizewell B nuclear power station (Figure 1) is sited on the East coast of England, near Leiston in Suffolk, adjacent to the Sizewell A twin-reactor Magnox station. It includes a single pressurised water reactor based upon an established reactor plant design from the USA. This is the first pressurised water reactor constructed in the United Kingdom for commercial power generation. The electrical output from the station at full power will be 1180 megawatts.

2. The Licensee applied for a nuclear site licence to construct Sizewell B in 1981. This licence was issued by the Nuclear Installations Inspectorate (NI) in 1987 following a lengthy public inquiry and submission by the Licensee of a Pre-Construction Safety Report. Table 1 provides a chronology of the key events during this period.

3. In granting the licence to construct Sizewell B, NI secured a commitment from the Licensee to complete a programme of further work, including the submission of a Pre-Operational Safety Report. NI also imposed a series of specific checks, or "hold points", on the stages of construction and commissioning. The Licensee could not progress beyond these hold points without the formal consent of NI (see Table 2). The final hold point related to the completion of commissioning and required NI consent for the station to commence full commercial operation. This consent has now been granted.

4. This report describes the regulatory process applied to Sizewell B and provides a summary of the activities carried out by NI in the period from the issue of the licence in 1987 to the granting of consent for full commercial operation in September 1995.

THE REGULATORY PROCESS

5. Under the Nuclear Installations Act 1965 (as amended), all sites used for commercial nuclear installations in the United Kingdom are required to be licensed by the Health and Safety Executive (HSE). The Health and Safety Executive has the power to attach such conditions to any nuclear site licence as may appear to be necessary or desirable in the interests of safety. NI, part of the Nuclear Safety Division of the Health and Safety Executive, has delegated responsibility for the licensing and regulation of such nuclear sites. NI is also responsible for enforcement at licensed sites of the Ionising Radiations Regulations 1985.

6. The nuclear licensing regime, in common with most UK health and safety legislation, is essentially non-prescriptive. That is, goals are set but the means of achievement are not prescribed. The licensee for each site is responsible for safety and for determining the means by which compliance with licence conditions is achieved. The means of achievement must be documented by the licensee. This requires the production of a safety case and arrangements for licence compliance. These are then subject to assessment and inspection by NI, to judge their adequacy (as described in subsequent sections of this report).
7. In exercising the licensing function, NII makes use of a number of controls derived from conditions attached to a nuclear site licence. These are consents, approvals, directions, agreements, notifications and specifications (Table 3 provides an explanation of these terms). NII has powers of enforcement under the Health and Safety at Work etc Act 1974 to issue prohibition and improvement notices and to prosecute for breaches of that Act or of the relevant statutory provisions.

8. Outwith the licensing regime, other regulatory bodies can have an involvement at licensed nuclear sites. The Field Operations Division (FOD) of the Health and Safety Executive is responsible for enforcement and inspection of non-nuclear safety matters. In England and in Wales (on behalf of the Welsh Office), Her Majesty’s Inspectorate of Pollution (HMIP) and the Ministry of Agriculture, Fisheries and Food (MAFF) are jointly responsible for matters of radioactive waste management policy and discharge authorisations. HMIP also has responsibility for certain environmental protection matters. Similar arrangements apply in Scotland. There are liaison agreements in place to ensure effective co-ordination between the various bodies.

9. In January 1981, the (then) Central Electricity Generating Board submitted an application to revise the nuclear site licence for Sizewell. At that time, the licence covered the existing power station, with Magnox gas-cooled reactors, and included provision for a second power station with an advanced gas-cooled reactor. The application made by the Central Electricity Generating Board sought to replace the advanced gas-cooled reactor with a single pressurised water reactor.

10. Following the application, a programme of safety submissions for the proposed new station was agreed which, when satisfactorily completed, would lead to the granting of a revised site licence. A significant item in this programme was the submission of a satisfactory Pre-Construction Safety Report for Sizewell B. Several drafts of this report were received and assessed by NII leading to the submission of the final pre-licensing version of the Pre-Construction Safety Report (Reference 1) in 1987. This document, together with a large number of supporting references, provided the safety case for all phases of the Sizewell B station (that is, design, construction, commissioning, operation and decommissioning) based on the state of knowledge at that time.

11. In some areas, work was sufficiently advanced to provide a definitive statement of the design and safety case, whilst in others research and development or analytical work was still in progress and hence the safety report contained a statement of the intended safety case together with a committed programme of further work needed to substantiate the case. After completing a thorough assessment of the Pre-Construction Safety Report, NII concluded that, although the safety case was not yet fully completed, there were no items of major significance outstanding which would preclude the issue of a licence.

12. In parallel with this assessment by NII, the safety case together with other aspects of the proposed station were subjected to scrutiny at a public inquiry which ran from 1983 to 1985 (Reference 2). On 12 March 1987, the Secretary of State for Energy granted consent under the Electric Lighting Act of 1909 for the construction
of the power station. NII granted a new licence for the Sizewell site on 4 June 1987, to include the pressurised water reactor (and to delete the provision for an advanced gas-cooled reactor - as in para 9). In doing so, NII had been satisfied with respect to the development of the safety case and had taken into consideration a number of additional items. These included:

- recommendations of Sir Frank Layfield’s report of the Public Inquiry;
- suitability of the site;
- review of the implications of the Chernobyl incident;
- establishment of a robust quality assurance system and a process for controlling post-licensing design changes; and
- agreement on a programme of post-licensing work.

13. The licence covers all phases of the station’s life from construction, through commissioning and operation to decommissioning. In granting this licence, NII took the opportunity to include a number of recommendations from the Layfield report in the conditions attached to the licence such as the requirement to develop and use a full-scale control room simulator for training operators. These recommendations became a legal requirement under the site licence.

14. In 1990, NII introduced a standard site licence for all nuclear power stations, with thirty five standard conditions. This change has not affected the principle or manner of regulation but has emphasised the Licensee’s responsibilities and the need for adequate arrangements to control activities important to safety (further details are given in para 287). Also in 1990, Nuclear Electric (NE) became the Licensee for Sizewell B on assuming responsibility from the Central Electricity Generating Board (CEGB) for nuclear power stations in England and Wales. For the remainder of this report, the term Licensee is used to encompass both CEGB and NE over the period from licensing in 1987 to 1995.

15. For the construction and commissioning phases of the project, the relevant licence conditions required the Licensee to make and implement adequate arrangements to control construction and commissioning. In making such arrangements, the Licensee divided construction and commissioning into a series of stages. The more significant of these were formally specified by NII as construction and commissioning stages under the relevant licence condition. Having specified these stages (and hold points), the Licensee was then required to obtain formal and legal consent from NII to proceed from one stage to the next. This provided NII with the mechanism for firm regulatory oversight of construction and commissioning, while at the same time ensuring the Licensee retained responsibility for safety. The granting of a consent to proceed from one stage to the next required NII to be satisfied that the safety case had developed satisfactorily and that all relevant commitments given at the time of licensing were being discharged satisfactorily. This process of reviewing progress on further development of the safety case and
progressively discharging commitments avoided the deferral of outstanding work until late in the project.

16. For the construction phase, the formal stages were as follows (Table 2 provides details of the relevant dates and consents):

- pouring of foundation mass concrete;
- first permanent structural concrete;
- mechanical access to radioactive waste building;
- installation of reactor pressure vessel;
- installation of reactor coolant pump support legs;
- commencement of pre-stressing of primary containment;
- commencement of full-scope simulator training;
- commencement of primary protection system functional testing;
- primary circuit hydrostatic test; and
- delivery of fuel to site.

17. The satisfactory clearance of each of these formal stages necessitated involvement by NII staff throughout the construction of the station. This took the form of both a frequent presence on site by a nominated site inspector, assisted by inspectors with specialist experience as appropriate, and the progressive assessment of formal submissions from the Licensee. Hence, throughout construction, NII staff have had continuous contact with the developments that have taken place and have satisfied themselves that each stage was completed in a satisfactory manner. Further details of the NII involvement during these stages are given in the section on Regulatory Inspection Activities.

18. For the commissioning phase, the formal stages were as follows (Table 2 provides details of the relevant dates and consents):

- stage 1 - initial test period (initial plant testing and cold functional testing);
- stage 2 - system and integrated testing period (unfuelled hot functional testing and preparations for fuel load);
- stage 3 - pre raise power period (fuel load and low power testing up to 5% power);
- stage 4 - initial raise power period (5% to 60% power); and
19. Each of these stages progressively demonstrated the satisfactory functioning of the station, commencing with plant completion testing on individual items of plant and culminating with the entire station. As for the construction phase, the Licensee was required to seek formal and legal consent before proceeding from one stage to the next. Before granting such consent, test results from the current stage were submitted to NII for assessment and only when satisfied with these results, together with the arrangements for the conduct of tests during the next stage, did NII grant consent to enable the Licensee to proceed. A requirement was also placed upon the Licensee to obtain NII agreement prior to taking the reactor critical for the first time. This occurred during, and was an additional control measure subsequent to consent for, Stage 3 commissioning (see Table 2). This staged process necessitated the involvement of NII staff throughout the commissioning phase, again through the nominated site inspector supported by colleagues as appropriate. Further details are given in the section on Regulatory Inspection Activities.

GENERAL WORKING ARRANGEMENTS

20. The NII work on Sizewell B involved the two assessment Branches (B and C) and the power reactor inspection Branch (E). In the period from 1987 (licence to construct) to 1995 (consent to start full commercial operation), some 107 staff years of effort have been expended on Sizewell B (see Table 4).

21. The nature and range of the work carried out during the post-licensing period necessitated extensive contact and interchanges between NII and the Licensee. A number of measures were applied, based upon pre-licensing practice, to ensure there were clearly defined levels and points of interface between the two organisations. This was important to the satisfactory control, monitoring and ultimate completion of the post-licensing issues and activities, particularly in view of the number and diversity of technical disciplines and work areas involved.

22. Specific programmes of work were established by the Licensee, and agreed by NII, to complete the necessary safety case submissions in the post-licensing period. These were known as Licensing Activity Summary Programmes (LASP). Separate programmes were drawn up for different work areas, of which there were fifteen in total (see Table 5). The content of each programme was agreed, and progress monitored regularly to achieve completion.

23. Meetings between NII and the Licensee took place within an agreed hierarchical structure, with designated Levels I, II, III and IV (see Table 6). The higher level (namely I and II) meetings involved senior management from NII and the Licensee, with representation also from Her Majesty's Inspectorate of Pollution (HMIP) and the Ministry of Agriculture, Fisheries and Food (MAFF). These types of meetings were generally held once or twice a year, or as necessary, to discuss matters of policy and the overall programming and progressing of work related to regulatory, activities. The lower level (III and IV) meetings took place more frequently, with less senior level of representation, to discuss technical matters and specific aspects of the regulatory process. Usually, attendance at Level III and IV
meetings involved only NII and the Licensee, although representatives of HMIP and MAFF frequently attended meetings on matters related to radioactive waste management. This enabled matters of common interest to be resolved in accordance with the agreed working arrangements between NII, HMIP and MAFF. One of the principal features of this hierarchical structure for meetings was the ability to refer upwards, for consideration at senior management level, matters which had not or could not be resolved at lower (working) level interfaces.

ASSESSMENT OF THE SAFETY CASE

Introduction

24. It is a requirement of the nuclear site licence that the licensee produces a safety case for all stages of the plant’s life, from design through construction and commissioning to operation and, finally, decommissioning. The format and content of a safety case is for the licensee to decide; it is not prescribed by NII. A safety case should comprise formal documentation which provides a comprehensive and cogent justification for the declared standards of safety and the means of achievement.

25. The Pre-Construction Safety Report (Reference 1), defined the main features of the design and safety case and provided the basis for the licensing of Sizewell B. In the post-licensing period, prior to operation of the plant, the Licensee had committed to complete the safety case based upon finalised design details and analytical work and to submit a Pre-Operational Safety Report. In addition, the Licensee committed to clear to the satisfaction of NII an agreed schedule, known as the 'D' Schedule, of confirmatory work or analysis identified at the time of licensing.

26. The Pre-Operational Safety Report (Reference 3) was issued to NII in November 1992. This report, together with supporting references, detailed analyses and design documentation, comprised the Licensee's safety case for the purposes of obtaining consent to load fuel in September 1994. Subsequently, the Pre-Operational Safety Report has been updated and superseded by the Station Safety Report. The latter will remain in place, being updated and amended as required, throughout the operating lifetime of the station.

27. The basis for assessment carried out by NII is provided by safety assessment principles published by the Health and Safety Executive. For the Sizewell B safety case, the NII assessment has been based upon the version of principles available at the time of licensing (Reference 4). Revised principles were published in 1992 (Reference 5) for application to future new designs of nuclear plant and facilities; it was not the intent to backfit the revised principles to existing designs. Nevertheless, a brief review of the Sizewell B design against the revised assessment principles in Reference 5 was undertaken. The earlier version of the assessment principles did not specifically address the use of computer software in safety systems. Therefore, the concepts embodied in the revised principles on this subject were applied to the Sizewell B assessment.

6
28. The Health and Safety Executive has also published its view on the tolerability of risks from nuclear power stations (Reference 6). This document is used by NII to form judgements on quantitative estimates of risk presented in safety cases.

29. In the following sections, the main elements of the Sizewell B design and safety case are described, with an explanation of the nature, scope and conclusions of the NII assessment. In carrying out the assessment, particular emphasis and effort has been applied to those areas considered to be of most significance to nuclear safety. This approach was based upon a knowledge of the design and safety case, experience gained from other nuclear plants and safety cases, and contact with national and international agencies and organisations. The NII assessment also specifically addressed the clearance of all items on the original schedule of post-licensing commitments.

30. The sections below are presented to generally correspond with the structure of the Licensee's Pre-Operational and Station Safety Reports.

Nuclear Safety Strategy

31. There are five fundamental principles which define the Licensee's strategy towards the achievement of a safe design, namely:

- no persons shall receive doses of radiation in excess of the statutory dose limits as a result of normal operation;
- the exposure of any person to radiation shall be kept As Low As Reasonably Practicable (ALARP);
- the collective effective dose equivalent to operators and to the general public as a result of operation of the nuclear installation shall be kept as low as reasonably practicable;
- all reasonably practicable steps shall be taken to prevent accidents;
- all reasonably practicable steps shall be taken to minimise the radiological consequences of any accident.

32. These are the same as the fundamental principles set down in the NII safety assessment principles. The fundamental principles embody the three main concepts of dose limitation, accident prevention and accident mitigation.

33. In the design of Sizewell B, to achieve these fundamental principles, the Licensee utilised its own design safety criteria (which embody the five principles above) and design safety guidelines. Areas where the design does not comply with the safety guidelines have been identified by the Licensee and reasons given why the non-conformances are acceptable.
34. Given that the Licensee's nuclear safety strategy is based upon fundamental principles in common with those set down by NII, the assessment carried out by NII concentrated upon the manner in which fulfilment of the principles was demonstrated in the appropriate areas of the safety case.

General Design Aspects

35. There are a number of general design aspects which have been applied either to the station as whole or to certain systems and equipment. These aspects include:

- Safety Categorisation of Structures and Equipment;
- Design Limits and Loadings for ASME III Mechanical Equipment;
- Equipment Qualification;
- In-service examination, testing and inspection including ASME XI requirements for safety classified components;
- Approach to Hazard Protection;
- Hazard Protection Implementation;
- Quality Assurance; and
- Achievement of Safe Shutdown

36. Each of these aspects is discussed, in turn, below. The NII assessment has encompassed consideration of the generic approach and specific examples of its application to plant or components.

Safety Categorisation of Structures and Equipment

37. The performance and integrity required of structures, systems and components is dependent upon their significance to nuclear safety. Consequently, every structure, system and component was evaluated during the design process to determine its safety significance. A comprehensive process of safety categorisation has been applied to ensure that consistent and appropriate standards of design, manufacture, construction and commissioning have been adopted.

38. The process is based upon three safety categories: safety category 1 encompasses equipment which forms the principal means of ensuring safety; safety category 2 encompasses equipment which makes a significant contribution to ensuring nuclear safety; safety category 3 encompasses all other equipment. Mechanical equipment in safety category 1 is subdivided into three safety classes (1 to 3) which relate more specifically to the functions performed by the equipment. Electrical equipment in safety category 1 is not subdivided into classes and is referred to as class 1E. Of particular importance, a specific category has been
introduced for a number of major components for which the possibility of disruptive failure without forewarning must be sufficiently remote as to be deemed incredible. These, designated incredibility of Failure (IOF) components, are the:

- Reactor pressure vessel shell and closure head;
- Steam generator channel head, tube sheet and secondary shell;
- Pressuriser shell;
- Reactor coolant pump casing;
- Reactor coolant pump flywheel;
- Reactor internals core barrel;
- Supports for the reactor pressure vessel, reactor coolant pumps and steam generators;
- Emergency core cooling system accumulator tank shells; and
- Main steam line no-break zone IOF pipework.

39. Civil structures have been categorised, in accordance with the equipment housed within them. Those structures which house the vast majority of category 1 and 2 equipment are allocated to safety category 1. The remainder are assigned to safety category 3 (category 2 is not used for civil structures).

40. A seismic categorisation process has also been applied to equipment and structures, based upon the required safety role to be performed during and subsequent to an earthquake. There are five seismic categories (1, 2, D, S and N). Equipment and structures which form the principal means of ensuring nuclear safety during or subsequent to an earthquake are placed in seismic category 1. Equipment which makes a significant contribution to nuclear safety following an earthquake is in seismic category 2. The supports for equipment which contains gaseous radioactive waste are allocated seismic category D. Equipment and structures whose failure during an earthquake could prevent equipment in seismic category 1 from functioning correctly, but which themselves are not required to remain operable or intact during or after an earthquake, are assigned to seismic category S. All other equipment and structures are placed in seismic category N.

41. Recognising the benefits of a structured categorisation process, the NII assessment has taken into consideration the category definitions, the standards applied to the various categories and the allocation of equipment and structures within the categories. As a result of this work, NII is satisfied that the Licensee has established and applied an effective categorisation process.
Design Limits and Loadings for ASME III Mechanical Equipment

42. As described above, equipment which forms the principal means of ensuring safety is designated safety category 1. For the majority of pressure retaining, fluid-carrying components in safety category 1 the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III (known as ASME III) have been applied. The code has been adapted for use in this country, to take into account differing institutional requirements between the United States and the United Kingdom. Equipment outwith the scope of ASME III has been designed to appropriate national or industry codes and standards.

43. The ASME III code sets design, service level and test limits for the evaluation of equipment and component design. For planned operations and for fault conditions which are sufficiently frequent to be considered within the design basis of the plant, the code requires a demonstration that components are capable of withstanding all applied loads with an appropriate margin. The code limits are set conservatively, with the highest degree of conservatism applied to the most frequent loading conditions.

44. Design to, and demonstration of compliance with, the ASME III code necessitates the identification of the loads on components which could arise from planned operations and design basis faults. Loads can arise from various sources (for example, from internal pressure and from external events). The nature and extent of the loads will vary according to the actual conditions on the plant. Therefore, it is necessary to establish these conditions and calculate the resultant loads on the components. This involves analysis of the way in which the plant behaves during operation (known as transient analysis). Under the ASME III code, the loadings and conditions used for design purposes are set down in the design specification for each component.

45. NII has reviewed and accepted the adaptation of the organisational requirements of the ASME III code for use in the design of Sizewell B. NII has also carried out assessment of selected transient analyses to confirm the validity of the approach and results, and of selected design reports which demonstrate that loadings on components are within the code limits. NII is satisfied that the loadings have been properly derived and analysed, and that the requirements of the ASME III code have been met.

Equipment Qualification

46. Safety related equipment needs to function correctly under operating and design basis fault conditions. To ensure this is the case, under anticipated environmental and seismic conditions, an extensive programme of equipment qualification has been carried out.

47. All equipment in safety category 1 and seismic categories 1 and 2 has been subjected to a formal process of qualification. This involved the determination of the conditions the equipment might need to withstand and demonstration that this could be achieved.
48. Environmental conditions encompass temperature, pressure, humidity, spray, dust, flood, chemicals and radioactivity. The conditions that equipment needed to be qualified against were determined by a number of factors, such as extremes in external ambient temperatures and conditions within the plant. Seismic conditions were determined from a defined level of earthquake and from the location and fixing of equipment. (The derivation of the earthquake appropriate to Sizewell B is discussed in the following sub-section on Hazard Protection, see page 13).

49. Equipment was qualified against the appropriate conditions by means of testing and/or analysis. In the case of testing, the required performance characteristics of equipment were verified before, during and after testing. Where applicable, equipment was artificially aged, to represent the condition at the end of expected life, before being subjected to testing. Qualification by analysis involved either specific consideration of the effect of the defined conditions on the equipment in question, or demonstration by equivalence. The latter method used evidence from the qualification or operation of similar equipment, under similar or more severe conditions, to obviate the need for specific further analysis.

50. NII has reviewed the overall programme of equipment qualification, taking into account international practice (particularly in the United States). Specific tests and analyses have been scrutinised to judge the adequacy of the applied methodology and confirm the acceptability of the stated results. Attention has also been given to the extent that ageing effects need to be taken into account in the qualification process. Overall, NII is satisfied that the manner and extent of equipment qualification for Sizewell B meets or exceeds best international practice, as applied to pressurised water reactors elsewhere in the world.

Hazard Protection

51. Hazards to the safety of the nuclear plant can arise from plant malfunctions or accidental events occurring within the Sizewell B site boundary and from events which originate from or occur outside the boundary. The former are termed internal hazards; the latter are external hazards. The significance of such hazards is that they have the potential to affect more than one component, system or safety feature at the same time.

52. To identify the hazards which needed to be specifically considered in the design and the safety case for Sizewell B, the Licensee carried out a comprehensive review of all potential hazards. Many hazards are of low probability of occurrence, or their effects are mitigated by design features on the plant. A number of hazards were identified as potentially significant on the basis of their possible effect or their probability of occurrence, namely:

- fire;
- pressure part failure;
- internal flood;
- missiles (arising from mechanical failures);
- earthquake;
- extreme environmental conditions; and
- miscellaneous hazards.

53. The miscellaneous hazards include dropped loads, aircraft crash and disintegration of a generating turbine. Extreme environmental conditions include wind, air and sea temperature, and lightning.

54. A level of severity was defined for each of these hazards, such that exceedance of the level within the operating lifetime of Sizewell B would be extremely unlikely. This established the design basis for hazards.

55. Protection against the defined hazards has been provided in one of several ways:

- by designing the plant to prevent or minimise the potential for occurrence of the hazard;
- by protecting the plant from the hazard, by means of a barrier;
- by designing the plant to withstand the effects of the hazard;
- by the provision of redundancy, diversity, segregation and separation of plant items;
- by providing equipment to mitigate the effect of the hazard (e.g., fire protection systems).

56. The licensee applied a set of general principles to develop the design of plant to withstand or mitigate the effects of hazards. From these, design criteria were determined for plant and equipment.

57. Those hazards (e.g., seismic and fire) which could contribute significantly to the frequency of an uncontrolled release of radioactivity have been included in the probabilistic safety analysis, which provides an estimate of the overall risk from Sizewell B (see the section on Fault Analysis, page 49). This has demonstrated that their contribution to the overall risk is small.

58. NII has carried out a comprehensive assessment which encompasses the most significant aspects and stages of hazard analysis and protection. This includes consideration of the completeness of the staged hazard review, the methods applied to defining and analysing the effects of hazards, and the adequacy of design measures to provide defences against specific hazards. Part of this work involved visits to Sizewell B to inspect plant layout and potential hazards.
59. Particular attention has been paid to the earthquake hazard, because seismic events have the potential to affect all parts of the nuclear plant and its safety systems simultaneously. This required the Licensee to first define the level of earthquake to be used for design purposes. This level is known as the safe shutdown earthquake (SSE).

60. The Licensee's original intent with respect to the seismic design of Sizewell B was that the station could be replicated at a number of sites within the United Kingdom, based upon a safe shutdown earthquake using a response spectrum with a peak horizontal ground acceleration of 0.25 g. The majority of structures and equipment were designed against 0.25 g peak ground acceleration.

61. At a later stage in the design process, the Licensee proposed the adoption of a safe shutdown earthquake specific to the site at Sizewell B. Considerable effort was applied to review the uncertainties associated with the data used in the derivation of the level of seismic hazard applicable to the site. The outcome of this work was the definition of a safe shutdown earthquake specific to the Sizewell B site having a peak horizontal ground acceleration of 0.14g.

62. In 1990, following further research and analysis, the Licensee determined that the predicted soil properties used in the design of Sizewell B needed to be revised. To ensure that this change did not invalidate the seismic qualification which had already been carried out, a reconciliation exercise was undertaken. The purpose of this exercise was to demonstrate that structures and equipment designed and qualified against a hazard level of 0.25g with the original soil properties were capable of safely withstanding a seismic event of 0.14g combined with the revised soil properties. The robustness of the design was further demonstrated by a review of the capability of selected structures and equipment to withstand a seismic event of 0.2g with the revised soil properties.

63. NII agrees with the choice of a safe shutdown earthquake with a peak horizontal ground acceleration of 0.14g, specific to Sizewell B. The results of the reconciliation exercise have been reviewed and are considered to be acceptable. The capability of the plant to withstand a seismic event is further demonstrated by the results of the assessment of selected items of plant at a hazard level of 0.2g. NII is satisfied that the predicted plant response, based upon the defined hazard level and revised soil properties, is likely to be conservative due to unquantified margins within the analyses.

64. Fire hazard is, in many cases, the bounding internal hazard. That is, the effects of fire and the measures taken to provide protection encompass (bound) and afford protection against other internal hazards. The provision of suitable and sufficient defences against fire (and the lesser hazards this encompasses) is another aspect of hazard protection which has been specifically assessed by NII. Past experience has shown that robust design for protection against fire hazards is essential. The design approach adopted by the Licensee conforms to the recommendations of the International Atomic Energy Agency and combines the elements of fire prevention, limitation of severity and limitation of the consequences of fire. This approach has resulted in a design which has a minimum amount of
combustible material and which has encompassed the principle of segregation of plant and equipment. The latter has been achieved either by the provision of physical barriers, with a design rating appropriate to the potential severity of the fire, or through separation by distance. Segregation also affords protection against other hazards bounded by fire.

65. NII has reviewed the measures taken to prevent or minimise the occurrence and severity of fire hazards. The adequacy of the design for segregation and of the provision of fire barriers have been considered in some detail and found to be consistent with best international practice.

66. Overall, NII is satisfied that the Licensee has applied a systematic and thorough approach to hazard identification, analysis, mitigation and protection which renders acceptable the contribution from hazards to the total risk from Sizewell B. Sufficient measures have been incorporated to achieve a design which is both robust and displays defence-in-depth against the identified hazards.

Quality Assurance

67. The overall quality assurance (QA) policy of the Licensee is to ensure that all activities, over the complete life cycle of Sizewell B - from design through procurement, manufacturing, construction, commissioning, operation and eventual decommissioning - are carried out in accordance with formal arrangements. The arrangements should meet the appropriate national standard for quality assurance and take into account best international practice.

68. The arrangements comprise a hierarchical documentation structure. There are QA programmes for each phase of the life cycle. These programmes cover the relevant organisational structure and responsibilities in place during that particular phase and describe the measures necessary to comply with QA standards. Under each QA programme there are lower-tier working procedures and instructions.

69. A process of QA categorisation has been applied by the Licensee. There are five categories, the highest of which applies to equipment and structures in safety category 1 (categorisation is explained on page 8). British Standards BS 5882 - Specification for a Total Quality Assurance Programme for Nuclear Installations - and the BS 5750 series have been applied as appropriate to the design, procurement, manufacture, construction, installation and commissioning phases of the plant.

70. In addition, the need for compliance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (Section III) for many of the components in safety category 1, with adaptation for application in the United Kingdom, influenced a number of elements in the QA arrangements. The main effects were the incorporation of specific arrangements for:

- the granting of a Certificate of Authorisation by the Engineering Inspection Authorities Board (EIAB) of the Institution of Mechanical
Engineers, as verification of the adequacy of the QA arrangements for specific components:

- the employment of Independent Inspection Agencies to witness inspections and tests;
- the certification by UK or European qualified engineers of design documentation required by the code;
- the use of specific proforma to meet code certification requirements;

71. To assess the adequacy of the Licensee's QA arrangements, NII adopted a top-down approach. The key top-tier QA documents were fully reviewed with selective, sample review of lower-tier documents. Assessment effort was concentrated in particular on the QA programmes covering the main phases of the project. In addition to the NII safety assessment principles and the requirements of BS 5882, the arrangements were judged against the International Atomic Energy Agency Code on the Safety of Nuclear Power Plants: Quality Assurance (50-C-QA). The arrangements were found to be adequate. Certain of the QA documents were formally approved by NII, under the terms of the nuclear site licence. Once approved, arrangements cannot be changed without the further approval of NII.

72. In order to monitor that the QA arrangements were being correctly implemented, NII carried out a series of formal audits. The areas selected for audit were wide-ranging and chosen to examine the various project phases, different technical aspects and the level of safety significance. No fundamental concerns were identified during these audits and all findings have been satisfactorily closed-out. Ad-hoc surveillance activities were also carried out, to supplement the formal auditing process. These surveillances examined quality assurance aspects in areas such as the production of design reports, the primary protection system software and independent assessment of documents. The adequacy of the resources applied to quality assurance by the Licensee, and the adequacy of the internal auditing procedures, were also monitored by NII.

73. The NII judgement, drawn from the above activities, is that satisfactory quality assurance arrangements, procedures and practices are in place at Sizewell B. It is a condition of the nuclear site licence that adequate quality assurance arrangements remain in place throughout the life of the station. NII will monitor compliance with this condition as part of the regulatory inspection programme for Sizewell B, throughout operation and eventual decommissioning.

Achievement of Safe Shutdown

74. A fundamental design requirement is that the reactor can be shutdown, either intentionally or as a result of the action of protection systems, with stable and safe conditions being achieved and maintained. Essentially, this requires means for controlling the nuclear reaction within the core and for providing adequate cooling to the fuel elements. The fuel elements continue to generate significant quantities of
heat when the reactor is initially shutdown, due to the action of fission products within the fuel matrix. This is known as decay heat.

75. Systems are provided to perform the functions necessary to achieve and maintain a safe shutdown. These include, for example, control rods backed up by a system for adding boric acid to the reactor cooling water (boron provides a means for controlling the nuclear reaction within the core) and a number of safety systems for cooling the reactor core in the event of a significant (albeit unlikely) leak of water from the reactor primary cooling circuit. Those systems which are provided to achieve safe shutdown conditions in the event of a fault on the plant are known as safeguard systems. The number, performance and reliability of safeguards systems need to be sufficient to ensure safe shutdown and reduce the risk of an uncontrolled release of radioactivity to as low as reasonably practicable.

76. The number and types of safeguards systems have been determined from consideration of those faults defined for the design basis of the plant. The adequacy of the safeguards provisions has been confirmed in the fault and transient analyses. The performance required of safeguards equipment and systems was determined from the fault analysis; tests carried out during plant commissioning have demonstrated the required performance can be satisfactorily achieved. Equipment surveillances will be carried out periodically throughout the life of the station to ensure the required performance levels are maintained. The equipment in safeguards systems has also been qualified, as necessary, against environmental and seismic conditions (equipment qualification is explained in more detail in another sub-section, see page 10).

77. The principles of redundancy, diversity and segregation have been applied to achieve sufficiently high levels of reliability for safeguards systems. To avoid reliance upon any single component or item of equipment, a design criterion has been applied such that systems must still be capable of performing the necessary protective function even with the failure of any single component. An extensive analysis of the design has been carried out to demonstrate that the single failure criterion has been met. Exceptions to the criterion have been justified on a case-by-case basis.

78. Nil has considered, in some detail, the adequacy of the safeguard systems. The fault analysis has demonstrated there are sufficient systems to achieve safe shutdown against the chosen design basis faults and the required performance has been demonstrated within the safety case and during commissioning of the plant. Particular attention has been paid to the application of the single failure criterion and the provision of redundancy and diversity. In addition to the design and commissioning measures, the Licensee's proposals for monitoring, inspection, testing and maintenance of safeguards systems have also been reviewed selectively and judged to be acceptable. The requirements for monitoring, testing, inspection and maintenance are set down in the technical specifications (see para 291); these also ensure there are sufficient systems available to achieve safe shutdown when any system is out of service due to maintenance or testing.
79. In summary, the Licensee has demonstrated that the design incorporates adequate provision to achieve and maintain safe shutdown. The extent to which diversity of systems and components has been included in the design is judged, by NII, to meet or exceed the standards applied to pressurised water reactors elsewhere in the world.

Human Factors

80. In the design of Sizewell B and the arrangements for its operation, extensive attention has been given to human factors - that is, all of the features which influence the performance of the personnel on the station. The objective of the human factors work is to support the personnel in their various roles and contributions to station performance, and to ensure that the risk of human error is reduced as far as reasonably practicable.

81. The design incorporates features which minimise the potential for, and the consequences of, operator error. This involved consideration of:

- the inherent characteristics of the design;
- automatic protection and controls;
- plant control interfaces;
- the main control room environment;
- monitoring, inspection, testing and maintenance;
- on-site emergency response facilities.

82. Human factors specialists within NII were involved in monitoring the development of the design and used established NII approaches, based upon previous experience and knowledge of human factors, to assess these facets.

83. The safety case for the station gives adequate definitions of the responsibilities and activities for safety across the range of station staff. This provides the basis for task analysis in support of the design and assessment of all human factors aspects. Indeed, there has been an increased emphasis placed upon task analysis. NII has confirmed that the task analyses, carried out during the whole development and construction programme, have covered an adequate range of contexts and application areas. Building on the task analysis work, the assessment of demands placed on personnel indicates that the required performance levels can be met, there are appropriate allowances of time for safety actions, the control room staffing arrangements are adequate, and the balance between operator action and level of automation is acceptable. A principal design objective for Sizewell B has been to minimise the necessity for operator actions, to protect or mitigate against faults, in the first thirty minutes following a reactor trip demand. This is known as the "30 minute rule".
84. The design process for the main control room and auxiliary shut-down room has incorporated the psychological and physical needs of the operating teams as specified in established human factors recommendations. This has been followed by comprehensive verification and validation. The computer-based display system has received appropriate human factors assessment by the Licensee, and the system supports the operators in their handling of alarms. The physical conditions and workspaces within the main control room and the auxiliary shut-down room meet established human factors standards and the adequacy of communication links on-site for emergency use has been confirmed by means of an analytical study.

85. The operating instructions have been comprehensively validated by a variety of task analysis studies, including trials on the control room simulator, and the standards of presentation and usability are very good. Similarly, the usability of the technical specifications has been confirmed by an NII team inspection. The technical specifications contain the administrative limits and conditions for all modes of plant operation. They also stipulate the safety-related requirements for monitoring inspection, testing and maintenance. The use of technical specifications is widespread elsewhere in the world, most particularly in the United States, but this is the first time they have been introduced in the United Kingdom. Details of the NII consideration of the acceptability of the Sizewell B technical specifications are given in the section on Regulatory Inspection Activities (see para 292). The emergency procedures based on critical safety functions are comparable with current international practice.

86. The selection of personnel for the station has been in accordance with the Licensee's arrangements and is satisfactory. A comprehensive training programme has been developed and undertaken and fulfils the requirements of the established systematic approach to training. The Licensee has gained considerable information from overseas pressurised water reactor operators, and training needs and training profiles have been defined in detail. A variety of training methods has been employed, including a multi-user basic principles training simulator and a technologically advanced full-scope training simulator. The latter was in use for training more than one year ahead of fuel load, as recommended by the Inspector at the Sizewell B public inquiry. Assessment of trainees, evaluation of training, and approaches to continuing training, have all been addressed to NII's satisfaction.

87. The safety case addresses management and organisation for safety. It reviews the station management structure, identification and division of responsibilities at both station and corporate level, selection and training arrangements, control of plant and procedure modifications, and arrangements for setting targets and monitoring performance. Some specific extensions of the review have been required by NII and have been accepted. The review includes also the consideration of safety culture and a comparison with principles derived from specific accident analyses.

88. Operator actions claimed in the fault analysis have been identified, modelled and quantified according to current human reliability assessment practice. The effects of recovery actions by operating teams and of interactions and dependencies
between actions have been included. NII accepts that a full quantitative treatment of errors of commission is not possible, but considers the qualitative assessment of such errors to be acceptable.

89. The judgement of NII is that the treatment of human factors issues in the safety case compares well with current best practice. The NII assessment confirms that all of the facets relevant to nuclear safety have been addressed, and that fully acceptable human factors standards have been met.

Reactor Core

90. The reactor core is the source of the nuclear heat energy which is converted, through steam generators and turbines, into electricity. The core of the reactor at Sizewell B utilises components, materials and technology which have already been proved in nuclear reactors in other countries (notably in the United States). The main components in the core are:

- fuel assemblies;
- rod cluster control assemblies;
- burnable poison assemblies;
- primary and secondary source assemblies;
- thimble plug assemblies.

91. There are [redacted] fuel assemblies. These are arranged into [redacted] different regions in the core, each of which contains fuel with a different amount (enrichment) of fissile uranium-235. The most [redacted] fuel assemblies are located in the peripheral region of the core. Each fuel assembly consists of [redacted] individual fuel rods, arranged in a square array (Figure 2). A fuel rod comprises a stack of uranium dioxide pellets, slightly enriched in uranium-235, enclosed in a tube manufactured from zircaloy-4.

92. There are 53 rod cluster control assemblies (RCCA), divided into two groups, for reactor control and reactor shutdown. Each assembly consists of 24 rods of a neutron absorbing alloy (silver-indium-cadmium) encased in a stainless steel tube. The burnable poison assemblies provide a means of reducing core reactivity in the early stages of operation, when the fuel is new and at its highest level of reactivity. Their effect diminishes with time; hence "burnable". The source assemblies emit neutrons to provide a base-line signal to the core monitoring instrumentation when the reactor is shutdown. Thimble plug assemblies are included to achieve the required distribution of coolant flow through the core.

93. The principal radiological hazard from a nuclear power station arises from the fission products and, to a lesser extent, the fissile material within the sealed fuel rods in the matrix of the core. Therefore, maintaining the integrity of the fuel and core assemblies is of primary importance. To meet this objective, the Licensee sets
down a range of functional requirements for the fuel and core components. A key requirement is that, under normal operating conditions or during fault conditions which could occur frequently, the fuel and core components should remain intact. Under certain extreme, infrequent fault conditions, where failures of the fuel cladding may occur, a further functional requirement stipulates that the reactor must still be capable of being shutdown safely and adequately cooled, to mitigate the consequences. Even under such extreme conditions, a radiological release to the environment should still be prevented by engineered barriers.

94. The Licensee has produced a safety case which encompasses the initial fuel design, yet provides scope for future enhancements to the design. The concept of safety analysis bounding limits has been adopted. This approach assumes certain upper limits or boundary values in the safety analyses for a number of fuel and core component parameters. Provided the bounding limits of these safety-related parameters are not exceeded by a given fuel design, the analyses remain applicable and the safety case is satisfied. In this way, a safety case has been produced which gives the Licensee flexibility in the choice of fuel and core component design for future cycles after the initial core load.

95. The NII assessment of the safety case specifically addressed the values used for safety limits, to establish that they were appropriate and bounding. The safety-related design criteria were also reviewed. No anomalies or omissions were identified and all limiting values were found to be acceptable.

96. The next stage of the assessment was to examine the case put forward to demonstrate the design meets the relevant criteria and safety limits. In a limited number of areas, it was found that design did not meet the necessary criteria. These areas, all of which had been identified by the Licensee, were considered in more detail. It was found that the safety analysis had adopted a conservative approach, using validated codes and methods and included allowances for uncertainties. For each of the identified areas, the Licensee made a reasoned argument to sustain the conclusion that fuel behaviour will be acceptable. NII considers this conclusion to be valid.

97. Other aspects specifically included in the NII assessment were the failure mechanisms associated with interaction between the fuel pellet and the zircaloy clad, and the in-core behaviour of previously failed fuel rods in any subsequent normal operation or fault situation. The Licensee had carried out a significant amount of work on both of these items and the safety case was found to be acceptable.

98. The analysis of the core design and performance requires the extensive use of complex computer codes. NII carried out a check to ensure that all codes declared by the Licensee were validated for the use described in the safety case. The use of one of the fuel performance codes (ENIGMA) was examined in detail. NII found that all codes were suitably validated and that the Licensee had followed sound practice in their usage.
99. In addition to the assessment activities described above, NII also made several visits to the fuel manufacturer. The purpose of these visits was primarily to seek information on, and discuss, detailed aspects of the fuel design.

100. The overall conclusion arising from the work carried out by NII is that an acceptable safety case has been presented, which demonstrates that the reactor core design provides an adequately safe source of nuclear heat. The risk to the general public from any potential radiological release meets the criteria given in the NII safety assessment principles and is considered to be as low as reasonably practicable.

101. Confidence in the performance and integrity of the reactor core is enhanced with the knowledge that there are no unique features or components employed in the design. These have already been successfully proven in reactors operating elsewhere in the world.

The Reactor Coolant System

102. The reactor coolant system (Figure 3) comprises the reactor pressure vessel and reactor internals, the pressuriser, four steam generators and four reactor coolant pumps plus interconnecting pipework, valves and instrumentation. There are four reactor coolant circuits, or loops, with one reactor coolant pump and one steam generator to each circuit.

103. The core of the reactor, containing the nuclear fuel, is housed in the reactor pressure vessel (Figure 4). This has a removable head, attached to the main body of the vessel by fifty four studs, to enable fuel to be loaded and unloaded. The four coolant circuits are connected to the reactor pressure vessel and the coolant water is pumped up through the core (Figure 5) to transfer the heat from the nuclear reaction in the fuel to the steam generators (Figure 6). The hot water in the reactor coolant circuits is used to generate steam, which is then used to drive a turbine to produce electricity (Figure 7). The steam circuit is separate from the reactor coolant circuit (the latter is often referred to as the primary circuit, and the former as the secondary circuit).

104. The pressuriser (Figure 8) is the principal component of the reactor pressure control system. To prevent the reactor coolant water from boiling, the primary circuit must be held at high pressure (over 2000 psi at normal operating conditions). This is achieved by maintaining a volume of steam in the upper part of the pressuriser, with water in the remainder. The pressuriser is connected to one of the reactor coolant circuits and primary circuit pressure is controlled by the pressure of the steam in the pressuriser. The steam pressure is regulated automatically by means of heaters and sprays within the pressuriser. There are two spring-operated safety relief valves and three pairs of pilot-operated relief valves attached to the pressuriser. These provide redundant and diverse means to protect the primary circuit from high pressure transients.

105. In addition to the main components described above, there are various safeguard and auxiliary systems connected to the reactor coolant system.
purpose of the safeguard systems is described in the sub-section on Achievement of Safe Shutdown (page 15), with further information in the section covering Engineered Safety Features (page 27). Auxiliary systems are discussed in the section entitled Other Systems (page 36).

106. The structural integrity of the reactor coolant system is fundamental to the safety of the nuclear plant. The components which form the reactor coolant pressure boundary are categorised as either IOF (see para 38) or safety category 1. They have been designed and manufactured in accordance with the ASME III code with additional UK requirements (as explained in the section on General Design Aspects, para 42).

107. The IOF components cannot meaningfully be subject to quantitative reliability analysis and specific measures must be taken to demonstrate that the failure of these components is incredible. This demonstration, known by NII as a special case procedure (Reference 5), requires several related but independent arguments to be used based upon:

- sound design concepts;
- the use of proven materials and components;
- the application of high standards of manufacture and inspection;
- rigorous quality assurance throughout all stages from design to operation;
- thorough testing during commissioning;
- the analysis of potential failure modes; and
- in-service monitoring of plant and materials.

108. The components should be as defect-free as possible. This high level of integrity is achieved by design, materials selection, high quality fabrication, inspection and testing. The integrity is demonstrated by analysis to show that the components are defect tolerant. The structural analysis considers the system operation and, in particular, includes severe transient conditions which can arise from extreme fault sequences. Considerable effort has been applied by the Licensee to the development of the structural integrity safety case, based upon these arguments. High standards have been met in the achievement and demonstration of integrity for these components.

109. All IOF components have been subject to appraisal by an independent third party inspection agency. This role has been performed for the Licensee by Lloyd's Register Industrial Division. Additionally, the stress analyses for the ASME design reports have been independently reviewed. The Engineering Inspection Authorities Board of the Institution of Mechanical Engineers has also been involved, as the body responsible for issuing to the Licensee the owner's certificate of authorisation.
The certificate signifies achievement of the necessary standards of quality assurance in organisation and management. The Inspection Validation Centre, independent of the Licensee and the fabricators, was established to validate non-destructive examination procedures and personnel. Lloyds, in their role as independent inspection agents, witnessed validation activities and ensured that the validated procedures were applied during works inspections.

110. The NII assessment of the structural integrity of the reactor coolant system has covered all the elements of achievement and demonstration of a high level of integrity. Particular attention has been paid to the IOF components and to the application by the Licensee of the measures set down in the special case procedure. This NII work has included:

- monitoring the manufacture and installation of components;
- selective review of design and fracture analysis reports;
- witnessing of component and system tests;
- review of inspection standards, techniques and validation for manufacturing, pre-service and in-service inspection programmes; and
- review of the operating and transient conditions defined for the reactor coolant system components.

111. For the reactor pressure vessel and other selected components (such as the casings for the reactor coolant pumps, the pressuriser and the steam generators) NII inspectors monitored their manufacture and installation. This was achieved by carrying out visits to manufacturers and to Sizewell B and by monitoring progress reports. These reports identified significant events in the manufacturing process, such as the need for weld repairs to components. Prior to and during installation of components at Sizewell B, NII confirmed that the necessary controls on cleanliness and contamination were in place. NII was satisfied with the standards of manufacture and installation.

112. ASME design reports for all IOF components and selected ASME class 1 to 3 components were subject to review by NII to confirm their adequacy as safety case documents and as lifetime records. NII also confirmed the reports had been produced in compliance with the Licensee's relevant procedures.

113. Fracture analysis reports for all IOF components were also reviewed by NII. The purpose of the fracture analyses was to calculate the maximum size of defect which could be present in a component at the end of its operating life, based upon the maximum size of defect likely to remain after inspection at the start of life and predictions of growth during service. The fracture analyses also calculated the size of defect which could cause catastrophic failure of a component. The ratio between the latter and the maximum predicted end-of-life defect size is known as the validation factor. This represents the margin of safety against potential failure.
Validation factors at, or approaching, two have been calculated for the reactor coolant circuit components.

114. At the main feed water nozzle region of the steam generators, a validation factor of about two can currently only be demonstrated based upon a restricted service life of twenty years. This is based upon the predicted defect growth during service. The Licensee intends to carry out additional in-service inspection of the main feed water nozzles to monitor defect growth and ensure adequate safety margins are maintained. From the assessment carried out, NII accepts that suitable validation factors have been demonstrated for reactor coolant circuit components. The results from the additional inspections of steam generator main feed water nozzles will be reviewed by NII to ensure that safety margins are not compromised.

115. Hydrostatic testing was successfully carried out on individual components including the reactor pressure vessel and the pressuriser; the tests were witnessed by NII. Subsequently, the hydrostatic proof test on the complete reactor coolant circuit was also witnessed. In this test, the reactor coolant circuit was filled with water and pressurised to 1.25 times the design pressure to verify the integrity of components, pipework and welds and to demonstrate there were no leaks in the circuit. A number of minor leaks were detected, at glands and seals, but none were of any significance and they did not invalidate the test. All the minor leaks were satisfactorily rectified during a later stage in commissioning.

116. NII has reviewed the inspection standards, techniques and procedures applied to components during manufacture (in-process inspection), prior to operation (pre-service inspection) and the standards and procedures to be applied throughout the operating life of the plant (in-service inspection). For components subject to the special case procedure, all inspections must be carried out to validated procedures. For in-process and pre-service inspection, there had to be redundant and diverse procedures. NII is satisfied that the Licensee has adopted suitable and validated procedures for the inspection of reactor coolant system components.

117. Similar procedures will be used for in-service inspections; there will be less redundancy of inspections due to access restrictions and the need to reduce radiation exposure to persons as low as reasonably practicable. The Licensee has developed proposals for in-service inspections and acceptance standards. These encompass the need to comply with the appropriate requirements of the ASME code, as adapted for use in the UK (see para 42). The proposals have been reviewed by NII and found to be generally acceptable. There are a number of aspects, such as specific acceptance criteria and the extent of inspection to be carried out on particular components, which still need to be finalised and agreed with NII. These will be kept under continual review by NII to ensure the details are finalised within a suitable timescale that does not compromise nuclear safety.

118. The Licensee has initiated a comprehensive surveillance programme to monitor the degradation of materials in the reactor pressure vessel during service. Capsules containing test specimens of the relevant materials have been inserted in the reactor. These will be removed in stages through the operating life of the plant.
and examined to determine the magnitude of degradation due to effects such as thermal ageing and irradiation embrittlement. NII has reviewed and agreed, and will monitor, the surveillance programme.

119. During the various modes of operation, or as a result of faults, the reactor coolant system will experience a range of transient conditions (essentially, combinations of pressure and temperature). These were defined in a schedule of design transients, which represented the operating "envelope" for the plant and components. This schedule has been reviewed and accepted by NII, with particular attention being paid to the more severe bounding transient conditions.

120. A specific fault has arisen on certain pressurised water reactors in other countries which has potential implications for Sizewell B and is being closely monitored by NII. Defects have been detected in some control rod drive mechanism penetrations, in the upper head of the pressure vessel, of reactors which have been in operation for some years. These penetrations are of the same type as those at Sizewell B. The Licensee has produced a programme of inspections to detect, at an early stage, any evidence of defects developing and will also continue to review world-wide plant operation and inspection experience. The Licensee's plans include provision for the manufacture of a replacement head, using improved penetration materials. This head would be installed should defects develop which might result in leakage. With these measures in place, there is no immediate requirement for replacement of the upper vessel head.

121. Elsewhere in the world problems have been encountered, primarily due to corrosion, with the material (inconel 600) used for the tubing inside steam generators. Following an extensive world-wide development programme, most new and replacement steam generators use tubing made from thermally treated inconel 690. The tubing in the steam generators at Sizewell B has been manufactured from this improved material.

122. The performance and reliability of components within the reactor coolant system and of associated auxiliary systems need to be commensurate with their significance to nuclear safety. The general design aspects (discussed in a previous section, page 8) such as safety and seismic categorisation, equipment qualification and hazard protection have been applied to these systems and components.

123. The NII assessment has considered the application of these general design aspects and the demonstration of acceptable levels of performance and reliability for particular systems and components. These include the reactor pressure control, the residual heat removal and the chemical and volume control systems. Equipment qualification and achievement of the single failure criterion were areas given particular attention. The provisions for in-service testing of components and systems, to confirm there is no significant deterioration in through-life performance, were also subject to review by NII. Experience of methods and practices applied in the United States, including liaison with the Nuclear Regulatory Commission (the regulatory authority), assisted in the assessment process.
124. For the assessment of the reactor pressure control system, NII commissioned a firm of consultants to carry out a detailed review for compliance with the single failure criterion. The equipment qualification carried out for the pressuriser relief valves was subjected to close attention by NII. There are two (diverse) types of relief valve namely, spring-operated single valves and pairs of pilot-operated valves. The same type of pilot-operated valve is in use on pressurised water reactors in France and the Licensee conducted a programme of qualification jointly with the French nuclear utility. Both types of valve underwent extensive testing; the results of these tests were reviewed by NII. In conclusion, NII judged that the reactor pressure control system is acceptable.

125. NII also utilised consultants to carry out a detailed analysis of the residual heat removal system (RHRS). This involved examination of the flow characteristics of the system, during normal operation and during transient or fault conditions. The work also encompassed a review of the acceptability of components for their selected duty. The results of the analytical studies on the system indicated satisfactory flow characteristics and performance would be achieved for the sample of flow conditions selected for consideration. During commissioning tests on the system, under conditions of high flow rates to simulate the response to a fault condition in the reactor coolant system, pipework vibration occurred, due to cavitation, which exceeded allowable design levels. A solution was implemented by means of relatively minor design changes, mainly involving the insertion of orifices in the system pipework, and proven by further testing. This illustrates the importance of commissioning testing to confirm the design intention and support performance substantiation within safety case analyses.

126. The chemical and volume control system (CVCS) performs several functions. These include controlling the reactor coolant water chemistry and the concentration of boron, providing a means for adding water to or discharging it from the reactor coolant circuit and injecting water into the seals for the reactor coolant pumps. There are a number of areas where compliance with the single failure criterion has not been achieved; principally failure of non-return valves or inadvertent closure of manually-operated valves. However, there are alternative systems available to perform the relevant safety functions in these cases and NII is satisfied that the design is adequate.

127. In summary, NII is satisfied that the structural integrity of the reactor coolant system has been properly achieved and demonstrated. Furthermore, the design, qualification and performance of the system, its components and related auxiliary systems has been shown to achieve the required standards. There are a number of aspects of the in-service inspection programme which need to be finalised. This work should be concluded within the initial phase of station operation. NII will monitor progress by the licensee to ensure timely and satisfactory completion.

128. The situation regarding the defects which have been detected in the control rod drive mechanism penetrations in certain pressurised water reactors elsewhere in the world will continue to be closely monitored. These defects do not occur until a reactor has been operating for several years. Sizewell B is a new reactor and inspections will be carried out to detect at an early stage any development of
defects. NII will review the results of these inspections. Therefore, it is NII's judgement that no immediate action need be taken with regard to replacing the reactor pressure vessel upper head at Sizewell B. If there are any developments which give rise to concerns over the integrity of the reactor pressure vessel head, NII will take appropriate action.

**Engineered Safety Features**

129. Engineered safety features are systems and components which are provided specifically to deal with faults on the plant. Many of these engineered safety features are actuated by the reactor protection system. Their principal functions are to ensure that the nuclear reaction is shut down, remains shut down and the reactor core is cooled following all faults, to prevent or minimise possible fuel damage. Engineered safety features are also provided for the reactor building to reduce radioactive releases to the environment following fault conditions involving leaks from the reactor coolant system.

130. Examples of engineered safety features include:

- emergency core cooling system (ECCS) which provides protection against loss of coolant accidents;
- emergency boration system (EBS) which provides a means diverse from the control rods for Shutting down the reactor;
- emergency charging system (ECS) which can provide boration and make up water to the reactor coolant system; and
- the auxiliary feedwater system (AFWS) for the steam generators which can maintain a means for reactor cooling should the main feed water system be unavailable.

131. The general design aspects (described in a previous section, page 8) such as safety categorisation of equipment, qualification and the single failure criterion have been applied to the engineered safety features. NII has carried out assessment of selected engineered safety features, taking into account achievement of the requisite general design aspects. NII has also carried out visits to component manufacturers and design contractors to examine their facilities and work undertaken on site. International experience has been taken into consideration to support judgements on the adequacy of the design and the in-service testing and monitoring arrangements.

132. One of the engineered safety features which has been subjected to assessment by NII is the emergency core cooling system (ECCS). This comprises several sub-systems which provide means for rapidly injecting large quantities of water into the reactor coolant system to protect against faults which result in significant leaks from the coolant system (these are known as loss of coolant accidents). There are three main sub-systems; two of which utilise pumps to inject
water into the reactor coolant system whilst the third utilises accumulator tanks pressurised with nitrogen gas.

133. The two sub-systems which utilise pumps perform complementary functions. The high head safety injection sub-system injects water at high pressure but comparatively low flow rate to protect against smaller leaks. The low head safety injection sub-system can deliver a high flow rate at low pressure in response to a large leak. The sub-systems draw water from a large storage tank, located outside the reactor building, which is filled with borated water. If and when the injection systems have drained the tank, the pumps can draw water from a sump at the bottom of the reactor building. The passive safety injection subsystem comprises four accumulator tanks, partially filled with borated water and pressurised with nitrogen gas. With the reactor coolant system at normal operating pressures, the discharge valves on the accumulator tanks are open; the presence of non-return valves effectively isolates the tanks from the reactor coolant system. In the event of a significant leak, which would cause the pressure in the reactor coolant system to fall below that set in the passive safety injection sub-system, the accumulators will discharge into the reactor coolant circuit (the non-return valves are positioned to allow flow in this direction).

134. NII, in the assessment of the emergency core cooling system, paid particular attention to the performance of the injection pumps and operation of the injection systems when drawing water from the sump in the reactor building. Liaison with the Nuclear Regulatory Commission in the United States was of assistance for this work. The application of the single failure criterion, including the effect of equipment not being available due to maintenance, was addressed and found to be acceptable. The provisions for in-service testing were reviewed and judged to be satisfactory; the Licensee has experience from the United States in deriving a suitable testing programme. The diagnostic equipment specified by the Licensee is in line with current best international practice; such equipment reduces the need for intrusive examinations of equipment and component disassembly. Data from the in-service testing and monitoring programme will be retained and reviewed by the Licensee to look for any adverse or unexpected trends in the results. This type of trending analysis can enable failures to be anticipated and appropriate pre-emptive action to be taken.

135. Structural integrity aspects of components in the emergency core cooling system have also been reviewed by NII; notably for the accumulators in the passive safety injection sub-system, which have been designated IOF components. NII assessed the ASME design report for the accumulators and the stress and fracture analysis reports. Although the derived validation factors are less than two for certain conditions, these are considered to be acceptable on the basis of likely conservatism in the fracture analysis and the nature of the load conditions in question (namely, infrequent events). For all normal loading conditions, the validation factors are greater than two. The accumulators will be subjected to a programme of in-service inspection which has been agreed by NII.

136. The emergency boration system (EBS) was another of the engineered safety features selected for assessment by NII. In general, the approach was similar to
that applied to the emergency core cooling system in respect of seeking an
acceptable demonstration of achievement of the general design aspects; particularly
equipment qualification and the single failure criterion.

137. This EBS system provides a diverse means of shutting down the nuclear
reaction in the reactor which is separate from the rod cluster control assemblies (as
described in the section on Reactor Core, page 19). The system comprises four
storage tanks, filled with a concentrated boron solution, connected via pipework and
valves to the reactor coolant system. In the event of any two or more rod cluster
control assemblies failing to operate correctly, the emergency boration system is
automatically activated by the reactor protection system. It can also be activated by
manual control. The boron solution in the tanks is injected into the reactor coolant
system, driven by the pressure head across the reactor coolant pumps. The
concentrated boron solution, when it flows through the reactor core, is an effective
neutron absorber and therefore shuts down the self-sustaining nuclear reaction.

138. Particular aspects of the emergency boration system which have been
examined by NII include measures to prevent dilution of the boron solution and its
effectiveness as a diverse shut down system. During commissioning tests on the
emergency boration system, excessive pipework vibration occurred. This was
diagnosed to have been caused by very high pressure pulses induced by the
fast-acting valves in the system. Modifications were proposed and agreed with NII
to rectify the problem. The principal changes involved an increase in the opening
times for the valves and some redesign of pipework and fittings. The effects of
these changes were analysed to ensure that the safety performance of the system
was not compromised; these revised analyses were reviewed by NII prior to the
modification being agreed. Tests on the modified system have demonstrated
excessive vibration no longer occurs. This case is a further example of the
importance of carrying out commissioning tests in support of design and safety case
analyses.

139. In summary, NII is satisfied that the design of the engineered safety features
is sound. In particular, the extent of equipment specification and qualification
carried out for Sizewell B and the extent to which diversity of safety systems and
components has been included in the design are judged to meet or exceed the
standards applied to pressurised water reactors elsewhere in the world.

Reactor Protection and Control Systems

140. The principal systems (Figure 9) which effect or contribute to reactor
protection and control are:

- the reactor protection system (RPS);
- the high integrity control system (HICS);
- the station automatic control system (SACS); and
- the station control rooms.
141. The reactor protection system monitors important reactor and plant parameters. On detection of abnormal conditions, the protection system initiates shut down (trip) of the reactor and activates the engineered safety features (the latter are discussed in a previous section). The system is fully automatic in its operation; no reliance is placed upon plant operators to initiate immediate protective actions.

142. There are two, entirely different systems which together comprise the reactor protection system. These are the primary and secondary protection systems (PPS and SPS, respectively). The primary system is fully computer-based, whereas the secondary system utilises long established hard-wired technology. The primary system provides protection against all faults within the design basis of the plant. The secondary system covers virtually all expected faults with respect to reactor trip and all but the least likely faults for the initiation of post-trip safety systems.

143. The primary protection system consists of plant sensors, multi-plexed signal processing, reactor trip and safety features actuation logic, output signalling, status communications diagnostics and integral auto-testers. There are four, segregated protection channels which operate on a “2-out-of-4” voting arrangement (Figure 10). If any two or more channels detect abnormal conditions the reactor protection measures will be initiated.

144. In other countries - for example USA, Canada and France - there are protection systems which depend in part or whole on software for their functioning. At Sizewell B, the primary protection system provides both reactor trip and post-trip functions, performed by a single integrated software-based system.

145. The safety demonstration required for the primary protection system was aligned to the special case procedure. This was designed, as used for certain reactor coolant system components (see para 107), strictly as a qualitative demonstration of fitness for purpose, there being no accepted means of quantifying the software reliability. It comprised the following elements:

- quality of production (including comprehensive checking and testing by the manufacturer);
- full independent assessment (examination and analysis by specialist teams independent of the manufacturer) devised to evaluate the quality of the software produced; and
- commissioning testing of the installed system (including independent assessment of test coverage)

146. The independent assessment element provided a searching examination of the integrity of the software, in a manner substantially diverse in nature from that provided by the manufacturer. The independent assessment encompassed:

- comprehensive manual checking of code and data;
• application of the static analysis tool MALPAS to the source code (aiming to achieve, where practicable, 100% code coverage);

• checking of the correctness of the object code (as installed on the system's programmable read-only memories) using a specially developed suite of software tools; and

• use of a specially developed dynamic testing facility, which applies a large number of randomly generated input combinations to the integrated hardware and software system.

147. The static analysis of all code amenable to MALPAS (approximately 86% of the total) was carried out by a firm of consultants. Their final report provides a clear summary of their findings; both the successes and the difficulties are informatively presented. Amongst the latter, the set of common functions software which manages the marshalling of communications within a master-slave processors' group (MSMIE - the multiprocessors shared memory information exchange software) proved not to be fully amenable to static analysis, because of its non-deterministic behaviour. The achieved code coverage for this software was about 80%. Confidence in the integrity of the balance of code, not amenable to static analysis, was derived from other sources of examination (principally the dynamic testing and extensive commissioning testing which the system experienced).

148. A number of the comments in the report from the consultants relate to detailed improvements which could be made to the software specification documentation. The comments have arisen because of the rigour in applying the MALPAS compliance analyser; if the natural language specification had not been of an adequate standard, the MALPAS analysis could not have been applied to the software. During the life of the Station it is inevitable that changes will be required to the software. Therefore it is considered essential for the documentation to be brought to, and maintained at, the highest possible standard. The contribution from the MALPAS exercise is of benefit in this respect. The Licensee has undertaken to review the comments in the report and to revise the documentation as appropriate.

149. It is apparent that the particular design of the software posed difficulties for the MALPAS analysts. The evident lesson here for the future is that, as far as possible, designers should take into account the tools available for analysing and testing their systems, as required for safety demonstration purposes, and tailor their designs accordingly. No software faults were found which would cause the loss of any protection function, but a significant number of non-critical anomalies was accumulated. About half of these became the subject of software changes. The overall conclusion of the consultants was that the statistics on the rate of comments per line of code are consistent with analyses performed on other safety-related systems of a similar age. NII has monitored the MALPAS exercise on a sampling basis. The work has been carried out thoroughly, professionally and to a high quality. NII judges the reported findings and conclusions to be valid.

150. The MALPAS work did not cover the (approximately) 100,000 lines of configuration and calibration data. Independent assessment of this data was carried
out by different consultants, using manual methods. In addition, pre-delivery and site commissioning tests were carried out. NII has reviewed the results from these activities and has concluded that fitness for purpose has been successfully demonstrated.

151. The object code has been systematically checked against its source for the full set of safety-critical software within the primary protection system. The task required the development of a special tool kit and was undertaken by the Licensee's Safety and Reliability Engineering Section. The checking has been repeated for each re-issue of the software. NII is satisfied that the work has been soundly executed.

152. Some anomalies were identified by the original analysis but none of a kind which would result in actual loss of a plant protection function. A potentially significant error was revealed within the Intel PLM66 compiler but shown not to pose a risk for the current software configuration. The later check analyses have followed a similar pattern of arisings and, together, have triggered a small number of changes (in the "desirable" category) to the software.

153. Dynamic testing of the software was achieved using a test harness which was a replica of a single protection system channel, with simulation of signalling from the other three protection channels. This enabled the software to be subjected to many thousands of randomised input challenges. An original test run of some 55,000 tests was performed. This encompassed scenarios within the band of the rarer protection demands (ie those not included within the designed protection coverage of the secondary protection system). A large number of "failures" was recorded amongst these tests and it was necessary to determine whether these were caused by the protection software or by shortcomings in the capability of the test harness. Following very extended and detailed analysis, the Licensee was able to demonstrate that in virtually all cases the cause lay with the harness.

154. There were a few exceptions, however, which could not be positively explained either in terms of the test harness or the protection software (although the former was suspected). The approach was taken, therefore, to implement planned improvements to the test harness and to re-run the suspect scenarios. The earlier observed failures did not (and could not be made to) recur, which was accepted by NII as reasonable confirmation that no safety-critical faults had been revealed in the original test run.

155. A further run of 5000 tests, using the (then) current version of the software, was subsequently carried out. This level of testing has been repeated for subsequent versions of the software, including that currently installed at Sizewell B. In all cases, for the full set of scenarios, some 10% (typically) of the tests produced anomalous results. This was a considerable improvement upon the original test harness performance but still left the Licensee with the task of proving a "clean run" for each version of the protection software. This has been successfully achieved for all software revisions to date.
156. Commissioning of the primary protection system at Sizewell B covered a period of some two and a half years, which included (and substantially exceeded) the targeted 12 months pre-operational soak test for the full system. The operation of the system, firstly as four separate channels and then as a fully integrated voting system, was tested by the commissioning team against the Licensee's original requirements specification. The connections to sensors and actuators were made in stages throughout the soak test period, and comfortably achieved the mean proportion of one third which had been considered appropriate for the viability of the test. The pre-operational phase culminated with the hot functional test for the plant as a whole. The primary protection system performed well during all stages of the scheduled commissioning.

157. NII is satisfied that the commissioning work was systematically and carefully carried out and that the system's functions have been adequately proven by test. A quarter of the commissioning query reports were raised specifically against the protection software and were dominated by specification interpretation errors. However, they embodied no pattern of findings to cast doubts on the basic integrity of the code itself. The observed rate of hardware (board) failures suggests that the operational reliability will be an order of magnitude better than originally predicted.

158. Although the various independent assessments, discussed above, found no code errors able to prevent the correct operation of the software, the findings nevertheless gave rise to a significant number of software changes. Additional changes were also identified from the site commissioning activities. In total, the verified software has experienced some six revisions since its initial installation on site. It is most important that an effective software change process has been applied. NII has assessed the procedures in considerable detail, in terms of both content and application. A key aspect examined has been the use of a formally recorded impact analysis for each change, to ensure that all potentially affected parts of the software are taken into account and to ensure adequate post-change test coverage.

159. Although formal records were not preserved for the impact analyses carried out in association with the early versions of the software, examples of software changes from this period have been examined by NII and indicate that a careful and systematic activity took place. Nevertheless, as a prudent measure, the Licensee was asked to carry out an additional review of the system's test coverage, to reinforce confidence that every function of the latest (operational) version of the software has been adequately proven.

160. Due to the lack of any established means for quantitatively establishing the level of reliability to be achieved from a software-based system such as the primary protection system, NII has accepted that the safety case would have to be developed without the support of a specific quantitative analysis. With the Licensee's application of an enhanced and particularly demanding depth of qualitative demonstration it was judged that explicit reliability quantification was not essential for the achievement of an acceptable safety case.
161. The Licensee has investigated, by means of the probabilistic safety analysis, the sensitivity of the overall risk from the plant to the reliability of the protection system. This entailed modelling of the relationship between protection system failure, the corresponding frequency of large uncontrolled radioactive release and estimation of the risk to an individual beyond the boundary of the site. The risk from the plant was calculated for each of a range of assumed values for the reliability of the primary protection system. The results showed that a level of reliability an order of magnitude lower than the originally specified target of $10^{-4}$ for probability of failure on demand (pf0d) could be tolerated for the primary protection system without significantly affecting the overall risk from the installation. The Licensee was also able to demonstrate more inherent defence-in-depth within the plant provisions than had been anticipated in the Pre-Construction Safety Report.

162. NII judges that the primary protection system is of acceptably high quality and is satisfied that an adequate demonstration of fitness for purpose has been provided. This is judged in the context of demonstrated defence-in-depth, backed-up by the secondary protection system and other plant and procedural arrangements and the risk-based sensitivity analysis.

163. The secondary protection system is constructed from traditional, hardwired electromagnetic (Laddic) technology. Its design is closely based on similar systems installed at Heysham 2 and Torness gas-cooled reactor power stations in 1987. These, in turn, were evolutionary developments of reactor protection systems which have been in use throughout the UK for more than thirty years.

164. The secondary protection system, like the primary protection system, comprises four segregated protection channels which operate on a "2-out-of-4" voting arrangement (see para 143). Laddic systems have been installed on all of the advanced gas-cooled reactors (AGRs) in the UK. To date, NII is not aware of anything which would cast doubt on the high levels of integrity claimed and expected from such designs.

165. Because of the evolutionary nature of this system, the NII assessment for Sizewell B has been concentrated primarily on the variations from earlier applications. The assessment has focused on:

- the correctness of the functional requirements specification;
- the quality of the manufacturing process; and
- the quality of installation and commissioning.

166. Examination of the Licensee's documentation, together with inspection of the equipment and attendance during key stages of site commissioning activities, have provided NII with sufficient evidence that the secondary protection system is fit for its intended purpose.

167. The control and instrumentation (C+I) facilities for the reactor and its auxiliary systems are provided by both computer-based and hardwired systems. The
computer-based systems consist of the high integrity control system (HICS), the process control system (PCS), the distributed control system (DCS) and the control rod drive system (CRDE). In addition, there is a set of controls and indications in both the main control room (MCR) and the auxiliary shutdown room (ASR) which are independent from the computer-based C-I systems.

168. The HICS system provides the majority of the station automatic control systems (SACS), namely, control of reactor coolant temperature, steam dump, main feedwater flow, steam generator level, pressuriser level and pressuriser pressure. The SACS are distributed between the same four segregation groups as the primary protection system (PPS) so as to reduce the potential for multiple control loop failure. In addition to the above control systems, HICS provides the high-integrity display of safety significant information (some of which is received from the PPS) in the MCR and the ASR, plus facilities to activate individual engineered safeguards equipment.

169. HICS employs some of the same computer hardware and software as the PPS, and is thus configured with dual redundant data highways giving increased fault tolerance. This common usage provides a high degree of assurance that the HICS system is of an integrity that is commensurate with its required duty. Also, NII is satisfied the Licensee has addressed the potential for HICS itself to initiate faults, and that it has been satisfactorily demonstrated that the reactor protection system has accommodated these faults in its design.

170. Particular attention has been given to the potential for common cause failures between the plant's control and protection systems. NII is satisfied that this potential is insignificant. On the question of the control systems' software reliability, the Licensee has demonstrated, by means of risk sensitivity analysis, that the requirement is modest; no more demanding than that considered achievable by proprietary industrial software. NII is satisfied that the control systems' software comfortably meets this standard.

171. The adequacy of the control and instrumentation provisions was demonstrated by the Licensee through an analysis which determined the operators' safety tasks and then ensured that sufficient control and instrumentation equipment was available to accomplish these tasks. In the event of a loss of facilities in the MCR (for example, due to a fire) a redundant set of facilities has been provided in the ASR which will enable the reactor to be brought to a safe shutdown state. The Licensee has satisfactorily assessed the adequacy of this facility and NII has had the opportunity to observe the successful change-over of control from the MCR to the ASR during a demonstration exercise.

172. The amount of HICS equipment required to be in service for continued operation has been acceptably established by the Licensee and incorporated into the station's procedures. NII accepts the Licensee has demonstrated that there are sufficient hardwired controls (i.e. independent from the computer-based facilities) to enable the reactor to be shutdown safely from the MCR in the event of a fault.
173. Control rod position is measured by a safety category 1 grade system (see para 38) which is separate from the control rod drive equipment (CRDE). Additionally, the PPS contains a number of checks on the alignment and positioning of these rods which protect against failures of the CRDE. Thus, NI1 is satisfied that the protection provisions in regard to the CRDE are adequate.

174. NI1 is satisfied that the Licensee has performed sufficient testing of the reactor control and instrumentation facilities to demonstrate their adequacy.

Main and Essential Electrical Systems

175. The safety function of the electrical systems is to provide power to any item of equipment or system needed to achieve safe shutdown of the reactor or otherwise to ensure nuclear safety. When the station is generating electricity, this is exported to the national grid via the main electrical system. Power can also be imported from the grid and the main electrical system distributes power, at the required voltages, to the station plant and systems (Figure 11).

176. The essential electrical system supplies power to safety related equipment and comprises alternating current (ac), direct current (dc) and uninterruptable power supply systems. The essential system is divided into four, electrically independent trains. The equipment and cabling associated with each train is segregated from the other trains. There are a number of exceptions to this principle with respect to cable runs; these have been justified on a case-by-case basis. Nuclear safety related systems are configured such that interdependent plant and equipment are supplied from the same train.

177. In the event of a reactor trip, when the station can no longer generate electricity, the preferred source of electrical power for the main and essential electrical systems is the national grid. Should grid supplies not be available, for whatever reason, there are four on-site diesel generators to provide electrical power to equipment required for safe shutdown of the reactor. The diesel generators are housed in two separate buildings, located on opposite sides of the reactor building (Figure 1). Each diesel generator is within its own compartment, has its own auxiliary equipment and fuel oil day tank and is aligned to one of the four electrically independent trains.

178. The diesels start automatically on loss of off-site power. There are also two smaller diesel generators which are provided for charging the batteries of nuclear safety related electrical systems should both the grid and main diesel generators fail simultaneously.

179. Generally, cables with reduced fire propagation characteristics have been used within the electrical systems. These cables also give low levels of smoke, toxic and corrosive emissions in the event of a fire. In certain areas, PVC cabling needed to be used. This use has been justified for each specific application. Cables have been routed to provide the necessary separation and segregation between various types and groups of cables. For example, power cables are physically separated from sensitive control and instrumentation cables. To mitigate
the effects of hazards, such as fire, certain groups of cables (and equipment) are segregated by means of physical barriers. The groupings are so arranged that sufficient equipment and cabling should be protected from a hazard caused by, or affecting, other areas to enable the reactor to be shutdown and maintained in a safe state.

180. To protect personnel and plant from electrical faults or lightning discharges, there is a station earthing and lightning protection system. The Licensee carried out an extensive study to examine the susceptibility of Sizewell B to the lightning hazard. For essential electrical equipment, the earthing system is arranged so that loss of earthing to one train of equipment does not affect other trains. Lightning protection is provided for buildings and equipment is protected from the effects of lightning. For example, where necessary, protection is provided against surge voltages on power supplies and equipment is screened against electromagnetic effects.

181. The NII assessment of the safety case for the electrical systems considered the adequacy of the system design under normal and fault conditions. This included a review of the Licensee's application of the single failure criterion and the provision of redundancy, diversity and segregation to ensure that, in the event of faults, sufficient essential electrical supplies could be maintained. In the review of the system design, NII also examined a sample of the electrical system design calculations to confirm the acceptability of the loads and duties assumed in the safety case. The response of the electrical systems to abnormal or fault conditions, the resultant fault levels and the reliability of protection devices were encompassed in the assessment.

182. NII looked at the types and specifications of cables. Details of tests carried out to demonstrate that cables had satisfactory characteristics with respect to flame propagation and smoke emission were reviewed, to confirm that the design and safety case requirements had been achieved. The Licensee's arguments for the use of PVC cables in specific areas were examined and found to be acceptable.

183. Other areas examined by NII during the assessment included: the routing, separation and segregation of cables; the integrity of cable penetrations through the primary containment boundary; the measures taken with respect to lightning protection; the capability of equipment to withstand hazards, and monitoring and maintenance of essential batteries. NII also gave consideration to the manner in which the electrical systems were commissioned and selected commissioning tests were witnessed.

184. The overall judgement of NII is that the electrical systems have been well designed, with sound application of the principles of redundancy, diversity and segregation. The required performance of the systems under normal and fault conditions has been demonstrated in the safety case, supported by commissioning tests, and shown to be acceptable.
Other Systems

185. In addition to the reactor coolant system, the engineered safety features and the electrical systems (discussed in previous sections) there are other station systems which are relevant to nuclear safety. These include:

- auxiliary cooling water systems;
- heating, ventilation and air conditioning systems;
- fire protection systems;
- overhead handling systems;
- the steam and power conversion system.

186. These other systems differ from the engineered safety features in that they do not provide the principal means of ensuring nuclear safety. They can, however, provide a contribution to safety. Many of the systems support normal operation of the station. Failure of these systems will not prevent the successful operation of the engineered safety features. Therefore, they perform a supplementary rather than a primary nuclear safety role.

187. Although the principal role of the steam and power conversion system is to generate electricity from the steam produced in the steam generators, using the heat produced in the reactor core, it also provides a means for removing heat from the reactor core under normal cooldown and certain post-fault conditions. Conversely, faults or failures in the steam and power conversion system, such as a break in main steam pipe or loss of feed water to the steam generators, could demand protective action from other systems to mitigate the effect on the reactor core.

188. Systems which have been selected by NII for assessment include:

- the reserve ultimate heat sink (RUHS);
- the reactor building polar crane;
- the main steam and main feed systems.

189. In carrying out the assessment of such systems, many of the general points discussed in previous sections were applied. These include consideration of design requirements, review of analyses and supporting information to demonstrate acceptable performance and carrying out visits to manufacturers and to the Sizewell site to inspect construction, installation and commissioning activities.

190. The reserve ultimate heat sink provides a means of removing heat from the reactor core which is diverse from systems reliant upon sea water cooling. The system comprises two duplicate and segregated sub-systems, each comprising a
heat exchanger cooled by air with associated pumps, valves and pipework. The system is seismically qualified. Specific aspects of the system examined by NII, and found to be acceptable, were the reliability analysis and protection against environmental and seismic hazards.

191. The reactor building polar crane is used to lift major loads during the refuelling programme; this includes removal and replacement of the reactor pressure vessel head. The crane can also be used for handling items of reactor plant and other equipment during maintenance periods. The crane can not be used when the reactor is operating and measures such as interlocks and strict procedural controls are in place to prevent use of the crane from hazarding reactor plant and equipment during shut down stages. NII has paid particular attention to the structural integrity of the polar crane and its ability to withstand the safe shutdown earthquake. NII is satisfied that the crane has been conservatively designed and manufactured and inspected to a high standard. The crane has also been subjected to an overload test prior to entering service. The polar crane does not need to remain operational during an earthquake but it does need to remain structurally intact and be capable of safely supporting any load it may be lifting at the time of the seismic event. Otherwise, failure of the crane or dropping of the load could result in damage to the nuclear plant and associated systems. NII has reviewed the seismic qualification of the crane. There is significant conservatism in the design and its ability to safely withstand the safe shutdown earthquake has been satisfactorily demonstrated.

192. The main steam system (Figure 12) provides the means for supplying the turbines with steam to generate electricity. The four steam generators are linked to the turbine hall by four main steam lines. Each steam line has a quick-closing isolating valve; there are also safety and relief valves to protect the system against over-pressurisation. The supply of steam to the turbines is controlled automatically and depends upon the turbine load demand. The main feed system supplies water to the steam generators, to be converted into steam. This system performs an important nuclear safety role, even when the reactor is shutdown and no electricity is being generated by the station, in providing a means for removing heat from the reactor core. There is also an auxiliary feed water system, as a back-up to the main system.

193. In view of the importance of the feed system, NII engaged a firm of consultants to carry out a detailed, independent analysis of the performance of the main feed system. This encompassed the analysis of water flow within the system, the effects of transient events and the potential for erosion and corrosion of pipework and components. The outcome of this work, in conjunction with NII's own assessment, confirmed that the main feed water system has been well designed and is capable of achieving the required performance.

194. NII examined particular aspects of the main steam system; notably the performance of the main steam isolating valves and the safety valves. This included the witnessing of tests carried out on the safety valves and consideration of in-service testing policy. The structural integrity of the main steam lines was subject to review by NII, especially the section of pipework in the no-break zone, which has
IOF status (see para 38). NII took into consideration the standards applied to
design, manufacture and inspection. The stress and fracture analyses for the
no-break zone pipework were also assessed. Overall, NII is satisfied that the main
steam system is well designed and has been shown to meet the required safety
standards for performance and integrity. The details of the in-service inspection
programme for the pipework need to be finalised. NII will progress satisfactory
completion of this work, in conjunction with the overall programme (see para 127).

Radioactive Waste Management

195. Radioactive waste at Sizewell B can arise from normal operation,
maintenance activities or fault conditions. The waste will be in solid, liquid or
gaseous forms and of low or intermediate level activity. The management of
radioactive waste by the Licensee must comply with policy in the United Kingdom
and discharges must be authorised under the Radioactive Substances Act 1993. In
England, and therefore for Sizewell B, authorisations under the Act for a licensed
nuclear site are issued jointly by Her Majesty’s Inspectorate of Pollution (HMIP) and
the Ministry of Agriculture, Fisheries and Food (MAFF). Authorisations contain
conditions on the disposal of radioactive waste, including limits on the quantity of
radioactivity that can be disposed of.

196. The radioactive waste management policy in this country, as laid down by the
Government, essentially stipulates that:

- low level liquid and gaseous wastes may be disposed of by
discharging into the environment subject to authorisation by the
appropriate regulatory authorities;

- low and intermediate level solid wastes should be disposed of as soon
as possible to avoid the provision of costly and extensive storage
facilities;

- high level wastes should be stored for a period of at least fifty years to
allow the internally generated heat to decay away.

197. In addition, the Government has set down objectives which are based upon
the system of dose limitation prepared by the International Commission on
Radiological Protection (ICRP). These are that:

- all practices giving rise to radioactive wastes shall be justified (ie. the
need for the practice must be established in terms of overall net
benefit);

- the radiation exposure of individuals and the collective dose to the
population arising from radioactive wastes shall be reduced to levels
which are as low as reasonably achievable (taking into account
economic and social factors);
the effective dose equivalent from all sources, excluding natural background radiation and medical procedures, to representative members of a critical group shall not exceed 1 milli-sievert in any one year (however, effective dose equivalents up to 5 milli-sieverts are permissible in some years provided that the total dose does not exceed 70 milli-sieverts over a lifetime).

198. The Licensee has a corporate strategy for the management of radioactive waste which encompasses the policy and objectives set out above. This strategy has been applied at Sizewell B. A fundamental element of the strategy is to design, construct and operate the nuclear power station to minimise radioactive waste arisings (this is discussed further in the following section on Radiological Protection, page 42).

199. Radioactive wastes at Sizewell B are collected, processed and treated in a purpose-built waste treatment facility, prior to storage in this facility or disposal from the site. The waste treatment facility is located adjacent to the reactor building (Figure 1).

200. There are two types of solid radioactive waste arisings, namely: low level waste and intermediate level waste. The low level waste is processed (for example by compaction, incineration or encapsulation) to reduce the volume and then packaged for disposal away from Sizewell B. Intermediate level waste will be stored at Sizewell B pending agreement on a national policy for such waste. There is currently provision to store the intermediate level waste arising from several operational cycles at Sizewell B.

201. The NII assessment of the strategy and safety case for radioactive waste management has been carried out in close liaison with HMIP and MAFF. One area, in particular, where such liaison was apparent related to the Licensee's original proposals for the treatment of intermediate level waste. Under these proposals, the waste would have been conditioned, by means of encapsulation, prior to storage on-site. Notwithstanding the adequacy (or otherwise) of the safety case in support of this proposal, the liaison between NII, HMIP and MAFF established that there was a more fundamental objection to the proposal. Conditioning of such waste could have precluded other options for disposal, pending the availability of a national disposal facility for intermediate level waste, and was judged not to have been demonstrated to be the best practicable environmental option. The Licensee subsequently revised the proposals for the treatment of intermediate level waste, such that they were acceptable to NII, HMIP and MAFF.

202. Another area examined in some detail by NII was the capability of the radioactive waste facilities to handle arisings from a possible "whole-circuit" decontamination process for the reactor coolant system. Although development of this process is continuing world-wide, the waste facilities at Sizewell B should have sufficient flexibility to cope with the arisings if such a decontamination process is carried out in the future. The Licensee has made a commitment to monitor world-wide developments in this field and provision exists to connect additional facilities should this prove necessary.
203. The capability of the waste treatment facilities to deal with the waste arisings from reactor plant fault conditions was also considered by NII. Although the facilities have not been specifically designed to deal with fault conditions, the Licensee has been able to show, to the satisfaction of NII, that waste arisings from design basis faults can be dealt with safely.

204. The NII assessment examined other key aspects of the safety case, such as the radioactive source terms data used in the design and analysis of the waste treatment processes and the balance achieved between doses to on-site personnel and to the general public. Commissioning tests for the waste treatment facilities were also reviewed by NII and selected commissioning activities were witnessed.

205. NII is satisfied that the Licensee has an acceptable strategy for the management of radioactive waste arisings from Sizewell B. The on-site waste treatment facilities are capable of handling, in a safe manner, the known and anticipated radioactive waste arisings. Radioactive discharges to the environment and waste disposal must comply with the authorisations granted jointly by Her Majesty's Inspectorate of Pollution (HMIP) and the Ministry of Agriculture, Fisheries and Food (MAFF). It was a prerequisite for NII agreement to first criticality of the reactor (see para 19) that the Licensee had obtained certificates of authorisation. There has been, and will continue to be, close liaison and consultation between NII, HMIP and MAFF on aspects of radioactive waste management.

Radiological Protection

206. Radiation exposures to on-site personnel and to the general public arising from normal operation and maintenance, testing, refuelling or inspection activities need to be as low as reasonably practicable (ALARP) and be within statutory limits. To meet these combined objectives, the Licensee set design targets for exposures which were well below statutory limits and adopted design practices which would minimise the sources of radiation and reduce the levels of exposure. For on-site personnel (operators) the design targets are: 10 milli-sieverts maximum annual dose; 5 milli-sieverts average annual dose; a collective annual target of 2.4 man-sieverts. The annual target for public dose is 170 micro-sieverts, from all exposure pathways.

207. The measures applied towards meeting the design targets included:

- the selection of plant materials to minimise the level of activated corrosion products in the reactor coolant circuit;

- the layout of plant and provision of shielding to reduce radiation exposures;

- the provision of remotely operated equipment for the maintenance and inspection of plant;

- the choice of a reactor coolant chemistry regime to minimise radioactive arisings.
208. In a pressurised water reactor, a significant proportion of the radiation dose to plant operators arises from activated corrosion products in the reactor coolant circuit. The main sources of these corrosion products are cobalt and nickel. The use of materials which contained cobalt or nickel was reviewed and, where practicable, alternatives selected to reduce exposure levels. For example, the material used for the tubes in the steam generators was changed from Inconel 600 to Inconel 690. The latter has a lower nickel content and improved resistance to corrosion. A reduction in the level of cobalt in the stainless steel used for reactor internals was another area where improvements were realised.

209. NII has assessed and accepts the strategy adopted for the review and selection of plant materials. The Licensee has given a commitment to monitor the results of world-wide research and experience on the replacement of the cobalt-based, hard-facing alloy stellite. Also, consideration will be given to the possibility of substituting zircaloy for Inconel 718 in the manufacture of fuel grids for future changes of fuel in the reactor.

210. The design and layout of plant to minimise radiation doses to operators was given careful consideration. A 1/40 scale model of the plant was used to review the design of active areas. Components have been designed to reduce or eliminate the need for inspections and to facilitate safe and expeditious handling during maintenance periods. Particular attention was given to reducing doses to operators during refuelling of the reactor. Design measures and operational procedures have been adopted to reduce exposure times. Local shielding has been provided where necessary.

211. The provision of remotely operated equipment, for the purposes of dose reduction, was systematically reviewed. Where practicable, taking into account best international practice, suitable equipment has been provided. For example, a multi-stud tensioning device for the reactor pressure vessel head and automated equipment for inspecting tubes in the steam generators will reduce operator exposures during maintenance periods. NII is satisfied that the Licensee has given due consideration to the provision of remotely operated equipment.

212. Permanently installed shielding contributes to the achievement of design targets for radiation doses to plant operators. Shielding requirements were defined, taking into account plant layout and occupancy factors, and analysed using computer codes which have been validated against experimental data and compared with international practice. The data for radiation source terms used in the analyses has been derived from international plant data. NII judges that the data and methods used to justify the adequacy of installed shielding are appropriate and that they have been suitably verified and validated.

213. An important part of the Licensee's safety case, with respect to radiation doses to operators, is the demonstration that the plant is capable of being operated in line with statutory dose limits and doses will be as low as reasonably practicable (ALARP). As part of this demonstration, the Licensee performed a detailed operator task analysis, using conservative data from foreign utilities operating pressurised
214. The Licensee’s health physics arrangements for the site are fundamental to the control of radiation doses to individuals. The arrangements were examined in detail by NII during the later stages of construction and commissioning. At each stage, the arrangements were found to be satisfactory. NII witnessed the commissioning task involving the loading of neutron sources into the reactor. This is a situation where high levels of radiation are present. The Licensee demonstrated satisfactory arrangements for the control of individual and collective radiation dose levels.

215. In the estimation of radiation doses to the public arising from the Sizewell site (ie both Sizewell A and B stations), the Licensee carried out a detailed analysis of all exposure pathways and identified groups of people (critical groups) most likely to be affected for each pathway. There are three pathways namely direct radiation (from sources of radiation within the boundary of the Sizewell site), gaseous radioactive discharges and liquid radioactive discharges. The assessment of the predicted dose levels to the public has been carried out by NII in close liaison with Her Majesty’s Inspectorate of Pollution (HMIP) and the Ministry of Agriculture, Fisheries and Food (MAFF). The discharge of radioactive wastes to the environment is regulated jointly by HMIP and MAFF. The Licensee has had to obtain authorisations from HMIP and MAFF for the discharge of gaseous and liquid radioactive wastes. These authorisations impose maximum permissible levels and require the use of best practicable means to minimise discharges so that public exposure will be as low as reasonably practicable.

216. The chemistry regime adopted for the water in the reactor primary coolant circuit has a direct influence on the generation of radioactive arisings from corrosion products and, hence, on the levels of exposures to operators and to the public. The chemistry regimes selected for the commissioning stages and for reactor operation have taken into account international experience and relevant research and development work. Similarly, the selection of a water chemistry regime for the secondary coolant circuit has taken cognizance of appropriate world-wide operating experience and recognised best-practice for chemical treatment has been adopted. The arguments to support the chosen chemistry regimes for both the primary and secondary coolant circuits have been assessed by NII and are considered to be acceptable.

217. Overall, the Licensee has demonstrated that radiation exposures to on-site personnel (operators) and to the general public arising from normal operation and related activities at Sizewell B are likely to meet the design targets and will be within statutory limits. It has also been satisfactorily demonstrated that the predicted levels of exposure are as low as reasonably practicable (ALARP).

Fuel Storage and Handling

218. The fuel storage and handling facilities provide means for performing safely all activities associated with the fuel route. That is:
• delivery and storage of new fuel;
• refuelling the reactor;
• removal and storage of irradiated fuel from the reactor; and
• final despatch of irradiated fuel from the station.

219. The provisions for fuel storage and handling at Sizewell B have been based upon a tried and tested design used at pressurised water reactor plants throughout the world. The design has been modified, where necessary, to comply with UK requirements.

220. New fuel is delivered to Sizewell B in purpose-built transport packages approved by the Department of Transport. On delivery, the fuel is unpacked, inspected and then transferred to the fuel storage pond where it will be held under water in storage racks, to await commencement of refuelling operations. The initial charge of fuel (loaded into the reactor in 1994) was stored in a dry condition, for which a separate safety justification was made.

221. When the reactor is refuelled, fuel assemblies are removed individually from the core and transferred under water to be placed in storage racks in the fuel pond (Figure 13). The replacement, new fuel assemblies are then transferred using the same route into the reactor core. All fuel handling is carried out under water, using remote handling equipment.

222. Fuel assemblies which have been removed from the reactor will be highly radioactive. Transfer and storage under water provides both cooling for the assemblies and shielding for the operators. The fuel pond incorporates a cooling system to remove the heat energy generated by the irradiated fuel assemblies and maintain the pond water at a stable temperature. There is initial storage space in the fuel pond to accommodate the irradiated fuel arising from the first five years of operation. In addition, there is storage space for new fuel (equating to one third of a core) and space to accept a complete core (to allow for in-service inspection of the reactor pressure vessel). Ultimately, the fuel pond has the capacity, with the addition of more storage racks, to hold the fuel from up to seventeen years of operation if the need arises. However, if the Licensee wishes to store more than five years’ worth of fuel, a further safety case will need to be submitted to NII.

223. For transport away from Sizewell B, irradiated fuel will be loaded under water into a heavily shielded flask. The flask will then be sealed and prepared for loading onto a transport vehicle (Figure 13). Licensee has yet to decide upon the type of transport flask to be used. Approval will need to be obtained from the Department of Transport before any movement of irradiated fuel away from the site can take place.

224. The safety case has addressed the design and operation of the fuel storage and handling facilities, potential fault and hazard conditions and protection against radiation hazards. Exposure to and release of radiation arising from fuel handling and storage operations have been examined to ensure the levels are as low as
reasonably practicable. In the handling and storage of fuel assemblies, it is important to prevent an accidental nuclear reaction. This is known as a criticality accident and can occur if two or more fuel assemblies are placed too close together and/or brought into contact with a moderator (eg water). A comprehensive review of potential criticality hazards has been carried out by the Licensee and necessary measures applied to prevent such an occurrence. These include: the location of stored fuel in racks which ensure a safe spacing between fuel assemblies; the movement and handling of fuel assemblies individually; the addition of boron to the water in the fuel pond and in the fuel transfer route; the routing of water pipes as far as practicable away from the new fuel delivery and preparation area.

225. The NII assessment of the safety case has considered the design and operational aspects of the fuel storage and handling facilities and selected supporting analyses. In terms of design and operation, the assessment looked at aspects such as the acceptability of the design concept (eg fuel storage racks with fixed spacings), the provision of adequate levels of reliability and redundancy (eg fuel pond cooling systems), the inclusion of appropriate safety features (eg mechanical interlocks), the consequences of plant failures (eg dropping of a fuel assembly) and the procedural controls for handling and storage operations. In terms of supporting analyses, NII gave particular attention to those relating to the cooling of irradiated fuel assemblies. For example, the analyses carried out to demonstrate that a fuel assembly stuck in the transfer system or dropped onto the floor of the storage pond would be adequately cooled were assessed by NII and found to be acceptable. The Licensee’s review of criticality hazards was also assessed and judged to be satisfactory. The design and procedural measures applied to avoid or reduce the risk of a criticality accident are considered to be acceptable.

226. Overall, NII accepts that the safety case demonstrates that fuel storage and handling operations can be carried out safely for both new and irradiated fuel. The Licensee will need to submit a further safety case to NII should the need or intent arise to store more than the initial provision for irradiated fuel arising from the first five years of operation.

Civil Structures

227. The main structures which make up the Sizewell B station (Figure 1) are:

- reactor building;
- enclosure building;
- auxiliary building and control building;
- fuel building;
- radioactive waste process and storage building;
- turbine house;
essential diesel buildings;
- auxiliary shutdown and battery charging diesel building;
- circulating water structures;
- reserve ultimate heat sink building;
- pipe and cable tunnels; and
- sea defences.

228. All structures have been categorised (using the process described in the section on General Design Aspects, page 8) according to their significance to nuclear safety. Design and construction requirements and standards have been applied which are commensurate with the categorisation. The safety roles performed by civil structures include:

- housing and supporting safety related plant;
- protecting plant from internal and external hazards;
- providing shielding against nuclear radiation; and
- preventing uncontrolled radioactive releases to the environment.

229. The structures must be capable of, and the safety case demonstrate that, the structures can perform the required role(s) and withstand all relevant loading conditions. Typical loads covered included construction (ie temporary) loads, normal operational loads and exceptional loads. The latter arise from internal hazards, such as dropped loads and pipe breaks, and from external hazards such as extreme winds and earthquakes.

230. The reactor building (Figure 14), in addition to housing and supporting the reactor and its primary coolant system, performs the role of the primary containment structure. The function of the primary containment structure is to contain fission products accidentally released from the reactor circuit and to minimise radioactive releases to the environment. The structure is designed to withstand the temperatures and pressures arising from faults which could result in a significant discharge of reactor coolant into the containment building.

231. The design, construction and commissioning of the primary containment structure was subject to review by an independent inspection agency (IIA). This agency comprised staff from Nuclear Electric, separate from the project group responsible for construction of Sizewell B, and members of Lloyds Register. The role of the agency was to assess the design reports for the primary containment structure and to inspect the construction and commissioning activities. The agency
was appointed by the Licensee, as part of the arrangements to produce a high-integrity structure.

232. An additional step taken, to demonstrate the capability of the primary containment, was the construction of a $\frac{1}{16}$ scale model of the structure. This model was used to validate the design and analysis methodology. Specifically, the model was tested to confirm that the containment structure could withstand internal pressures in excess of twice the design requirement. This compared well with the results from predictive analyses.

233. When the construction of the reactor building had been completed, the primary containment was subjected to an internal over-pressure test (at 1.15 times the design pressure). Tests were also carried out at the internal design pressure to confirm that the containment was essentially leak tight.

234. The NII effort applied to the civil structures gave particular emphasis to the integrity of the reactor building and primary containment. The construction was subject to a number of hold points, wherein NII consent was required to proceed to the next stage of construction (see para 16). The purpose of these hold points was to ensure that all necessary work had been satisfactorily completed, and relevant documentation issued, before construction was allowed to progress to a further stage.

235. NII carried out assessment of design reports for the containment structure, along with other submissions linked to the various construction hold-points. All assessment comments and queries were satisfactorily resolved with the Licensee. Regular visits were made to the site, to inspect construction activities, and to manufacturers to examine the fabrication of components such as containment penetrations and pre-stressing tendons.

236. The methodology and criteria for the pressure and leakage tests to be carried out on the primary containment structure were reviewed and agreed by NII. Both tests were carried out satisfactorily and achieved successful results. The pressure test was witnessed by NII.

237. NII held regular meetings with the independent inspection agency for the primary containment structure. At these meetings, the work and findings of the agency were examined and discussed. It is the view of NII that the independent inspection agency carried out its duties very effectively.

238. Although considerable effort was devoted to the primary containment structure, NII also reviewed other safety significant civil structures to establish the adequacy of their design and construction. The activities involved were similar to those described above for the primary containment structure, namely assessment of design and analytical reports coupled with visits to inspect construction and fabrication processes.

239. Selected analyses and calculation packages were assessed to confirm that the conclusions of the safety case had been adequately substantiated. These were
found to be acceptable. Specific technical aspects which were considered by NII included the cement content necessary to resist ground water sulphate attack and concrete behaviour at high temperatures. These, and all other items considered by NII, were satisfactorily concluded.

240. General inspection activities carried out by NII, during construction of safety significant structures, involved such matters as non-conformances with specifications, reinforcement coupler strengths and concrete cube strengths. Also, an audit was carried out to establish the adequacy of the construction procedures and controls. This audit focused on the Auxiliary Building and provided confirmation that the procedures and controls were acceptable.

241. NII is satisfied that the safety significant civil structures at Sizewell B have been designed and constructed to appropriate standards and procedures. The necessary performance and integrity of the structures has been suitably demonstrated in the Licensee’s safety case. Particularly notable is the level of effort applied to achieve and demonstrate the required integrity for the primary containment structure.

Fault Analysis

242. The purpose of the fault analysis is to demonstrate that the nuclear plant is able to cope satisfactorily with a wide range of potential faults. The results from the fault analysis are used to demonstrate that the risk to the general public arising from potential faults in Sizewell B is very low and that there are no reasonably practicable means by which the risk could be further significantly reduced.

243. In totality, the fault analysis is an extensive package of work which commences with the identification of initiating faults arising from plant failures or other hazards and culminates in identification of possible plant damage states, resultant radioactive releases and the corresponding level of risk to an individual member of the public.

244. The Licensee’s overall design target for Sizewell B is that the risk of death to any individual member of the public should be less than one in a million per reactor year. This is consistent with Reference 6, wherein the Health and Safety Executive argues such a level of risk is broadly acceptable in comparison with the other risks that people run.

245. In the consideration of initiating faults and fault sequences, the concept of “design basis” has been applied. Design basis faults are those events which the plant has been designed to withstand, by means of safeguard systems which prevent or mitigate against these potential faults and maintain the plant within safety limits. Faults or fault sequences which fall outwith this criterion are therefore beyond the design basis. Although these could result in plant safety limits being exceeded, they are very low frequency events for which the provision of specific or additional safeguards is judged to be not reasonably practicable. The estimate of the overall risk from Sizewell B is the aggregate of the minor consequences attributable to design basis faults and the larger consequences arising from faults.
beyond the design basis, taking due account of the predicted frequencies of both types of faults.

Probabilistic Safety Analysis

246. A probabilistic safety analysis (PSA) is an integral and substantive part of the overall fault analysis. This provides a systematic means of identifying faults and fault sequences, their effects and their likelihood of occurrence. The analysis carried out for Sizewell B has three levels. The first (Level 1) estimates the frequency of plant damage sequences which could result in a release of radioactivity (for example, damage to the reactor core and sequences which result in bypass of the reactor building containment boundary). The Level 2 analysis, using the results from Level 1, determines the frequency of an uncontrolled release of radioactivity to the environment. The consequential risk of death to an individual member of the public is then estimated in the Level 3 analysis, together with estimates for a range of measures to establish the risk to society as a whole (societal risk).

247. All three levels of the PSA have been assessed by NII. The assessment encompassed the methodology and the detailed analyses. To assist in this work, NII employed a firm of consultants from the USA. They were chosen for the work because of their experience with and knowledge of PSA methodologies and applications world-wide. The results of the PSA were judged against the numerical targets given in the NII safety assessment principles. The adequacy of the case put forward to demonstrate the risk posed by Sizewell B is as low as reasonably practicable was also scrutinised.

248. The first stage of the PSA is a fault schedule which lists all of the initiating faults and hazards which could lead directly, or in combination with other failures, to plant damage. The Licensee's starting point for identification of initiating faults was to define categories of fault and then to determine faults within these categories using operating experience of similar plants and failure modes and effects analysis for the operating systems.

249. NII's key requirements were that the fault schedule should:
   
   • be as complete as possible;
   
   • cover all the modes of operation of the plant; and
   
   • include faults which could lead to releases from all of the sources of radioactivity on the plant.

250. NII recognises that completeness is a difficult attribute to demonstrate and the assessment was based to some extent on the rigorous and systematic approach employed by the Licensee to derive the fault schedule as well as on comparison with fault schedules for other plants. NII judges that all three requirements were met and the fault schedule includes a wider range of initiating faults and hazards than is normally covered in a PSA. The wider range of initiating faults has led to a better
overall estimate of risk and to a better indication of the relative contributions to the risk from Sizewell B.

251. A schedule was produced to identify the protection systems required to operate for each of the initiating faults and hazards. This “safeguards schedule” is very complex. Not only does it define the protection requirements for each fault and event group, it takes account of any equipment which could be rendered inoperable by the initiating fault or hazard and possible failures of parts of the protection subsequent to the fault. The safeguards schedule thus contains variations of the success criteria for each fault sequence as well as information on the availability of systems which may be called on during the fault. The actual success criteria used in the PSA have been supported by transient analysis.

252. It is NII’s judgement that the safeguards schedule gives a very detailed specification of the success criteria of the safety systems in fault conditions. The consultants employed by NII considered that this was a more complete specification of the success criteria than is normal in PSAs.

253. The sequences of events which could arise from the initiating faults on the fault schedule were analysed using event trees. The event tree analysis performs a number of functions in the PSA. Firstly, it is used to identify the fault sequences which are within the design basis. The definition of the design basis is extensive in that it includes all fault sequences down to a frequency of $10^{-7}$ per year. This includes all initiating faults and often multiple additional failures which could occur. This is much more extensive than the design basis for other plants which are typically defined deterministically - that is, the design basis includes the initiating fault with one additional failure. Transient and radiological analyses are carried out for the design basis sequences which set out to show that the plant would be brought to a safe, stable shutdown state if the safety functions are met, and that any release associated with these sequences remains within prescribed limits (see para 262). NII judges that the event tree analysis provides a very detailed and comprehensive definition of the design basis of the plant.

254. As well as “success” states, the event trees also identify failure sequences which would lead to reactor core damage or to a significant release of radioactivity, as a result of failure to satisfy those safety functions modelled in the event trees. The frequencies of these failure sequences have been quantified as part of the analysis. NII is satisfied that the event tree analysis correctly determines the frequency of the failure sequences for the safety functions modelled.

255. Functional fault trees were used to identify the fault sequences which lead to plant damage as a result of failure of those safety functions not already covered in the event trees. The functional fault trees model the failure of safety functions; each safety function can be performed by more than one system. Important inputs to the functional fault trees are the system fault trees which model how safety systems can fail. Bringing the system fault trees together in a functional fault tree ensures that the analysis takes full account of functional dependencies between different systems. This does, however, cause the functional fault trees to become very large and unwieldy making each analysis a very time consuming task.
256. Because of their size and complexity, some errors in the construction of fault trees were inevitable. Consequently, the trees themselves, together with the results have been extensively reviewed by the Licensee. During the assessment process, NII detected a number of errors, but gained confidence from the fact that the Licensee’s review process had identified and rectified these faults without prior notification, or had demonstrated the errors were insignificant. NII is satisfied that the fault tree analysis provides adequate estimates of the frequency of plant damage for the safety functions modelled.

257. The results of the Level 1 PSA are given in terms of plant damage states (PDS). Each PDS represents a group of accident sequences which are judged to have similar characteristics with respect to accident progression. They contain contributions from all of the identified faults and hazards, and from all sources of radioactivity on the plant. There are four basic PDS groups: core damage, reactor building damage, reactor building bypass and ex-reactor damage. The main result from the Level 1 analysis is an estimate for reactor core melt frequency of $3.32 \times 10^{-5}$ per year. NII judges there to be a degree of conservatism in the calculation and the estimated frequency is acceptable.

258. Plant damage states (PDS) provide the interface between the Level 1 and Level 2 PSA. For the core damage and the reactor building damage states, the magnitude of the release to the environment depends upon the ability of the reactor building to retain radioactive material. This is modelled in the Level 2 PSA using containment event tree (CET) analysis. The CETs model the phenomena that would occur, including reactor pressure vessel failure, steam explosion, hydrogen burn and containment failure. Probabilities are assigned to the nodes of the CET and the probabilities of the sequences are calculated. The end points of the CET are grouped into source term categories and assigned to appropriate release categories to determine the off-site consequences. NII is satisfied that the structure of the CET and the nodal probabilities are an adequate representation of the phenomena which would occur following a core melt sequence. The assignment of source term categories to release categories has been carried out in an acceptable manner. The results of the Level 2 analysis give a frequency of an uncontrolled release at $2.98 \times 10^{-5}$ per year. This is significantly higher than the objective of $10^{-6}$ per year, albeit the results are based upon conservative calculations. Hence there was a need for the Licensee to present a good justification that the risk is as low as reasonably practicable (ALARP).

259. From the Level 3 analysis, the estimated risk of death to an individual member of the public arising from fault conditions is $1.9 \times 10^{-7}$ per year. This value, when combined with the individual risk of death associated with normal operation ($6.9 \times 10^{-7}$ per year) is less than the target figure of $10^{-6}$ per year. In addition, the Licensee has given estimates for a range of factors affecting societal risk. These include the numbers of fatal and non-fatal cancers, and land contamination. Although there was no target for these societal risk measures, the results have been used to argue that there are no “cliff edge” effects at frequencies just below the target level for individual risk. NII judges that the Level 3 analysis is acceptable and agrees that there are no cliff edges.
260. In consideration of the requirement to demonstrate that the risks from Sizewell B are as low as reasonably practicable (ALARP), the position taken generally by the Licensee was that, if the risk of death to an individual off-site could be shown to be less than $10^{-6}$ per year, then the ALARP demonstration requirement had been met. In support of this general approach, ALARP has been considered for the plant damage states which give a significant contribution to individual risk and, at the request of NII, to the dose-frequency relationships, core damage frequency and the large, uncontrolled release frequency. In carrying out the ALARP investigation, the Licensee undertook a significant revision of the analysis, removing some of the conservatisms and taking credit for the adoption of accident management measures and the provision of additional monitoring equipment. The conclusion drawn by the Licensee is that there are no further reasonably practicable changes which can be made to the design or operation of the plant. NII has reviewed the ALARP investigation and accepts the Licensee's conclusion.

261. A programme of work to develop the probabilistic safety analysis into a "living" operational tool has been agreed between NII and the Licensee. This will enable the analysis to be updated more readily to take into account any future changes to the design or operation of the plant before such changes are implemented. Plant data will also be collected and used as a means of validating and updating the fault analysis. The programme of work also includes comprehensive sensitivity studies, to provide a further demonstration of the validity of the results of the analysis. NII will monitor and review the progress and outcome of this work.

Transient Analysis

262. Within the overall fault analysis, the purpose of the transient analysis is to demonstrate that plant safety limits are not exceeded. This demonstration is based upon representative sequences of events which stem from the design basis faults identified as part of the probabilistic safety analysis. Essentially, the transient analysis confirms the assumptions that the plant remains within the design basis for the postulated faults.

263. The ability of the nuclear plant to tolerate faults and fault sequences within the design basis has been assessed against five sets of plant safety limits. These are:

- radiological release;
- fuel rod structural integrity;
- primary circuit integrity;
- secondary circuit integrity; and
- containment integrity.
The first of these is a basic limit on radiological release. This determines that a fault within the design basis must not lead to a radiological release with off-site consequences greater than specified levels. There are in fact three separate bands for the radiological release limit, with progressively lower limits on release for higher frequency faults.

The remaining four sets of limits relate to the integrity of the physical barriers to the release of radioactivity. There are fault sequences which could exceed the plant safety limits, yet would not lead to failure of one of the physical barriers because the design has built-in safety margins which means it is capable of withstanding the effects of such faults. These fault sequences are identified by a process of design capability analysis. This enables a more realistic estimate of the frequency of radioactive releases to be derived, by distinguishing between those sequences which exceed safety limits without causing physical damage to the barrier and those which lead to damage and could result in a release.

Fault sequences which could challenge safety limits are identified using event tree analysis. The initiating faults, for consideration in the event tree analysis, stem from the fault schedule. There are a large number of initiating faults, which result in some 5000 design basis fault sequences. As it would not have been practicable to carry out event tree analysis for so many faults, the Licensee applied a process of reduction by allocating faults to event groups and then identifying a set of faults within these groups which bounded all others. These are called bounding limiting design basis faults (BLDBF) of which there are approximately 90 in number.

The bounding limiting design basis faults were divided into two groups for analysis; those considered to be bounding for the four physical integrity plant safety limits, and those which were bounding for the radiological release limit. For the former, transient analysis was carried out to demonstrate that the relevant limits would not be exceeded. For the latter, transient analysis results were used as an input to perform a radiological analysis to estimate the magnitude of any release.

The adequacy of the transient analysis for the faults bounded by the four physical integrity plant safety limits was assessed by NII, with three main areas for consideration:

- Were the computer codes used for the transient analysis properly validated and verified for the purpose?
- Was the input data for the analysis acceptable?
- Were the results of the analysis within the limits defined?

The BLDBF's for which transient analysis was performed can be broadly divided into two classes, namely loss of coolant accidents (LOCA's) and intact circuit faults. The primary and secondary circuit thermal-hydraulic behaviour resulting from these fault classes are very different and therefore fault-specific computer codes were required to determine the transient behaviour. In addition, certain BLDBF's result in the discharge of fluid to the reactor and auxiliary buildings
and it was therefore necessary to calculate the thermal-hydraulic conditions arising from these faults. In view of the importance of the code calculations the NII devoted significant effort to the assessment of the adequacy of the validation and verification for the main computer codes. As a result of this assessment, NII is satisfied that the main codes are fit for the purpose of calculating the thermal-hydraulic behaviour of design basis faults.

270. NII reviewed a selection of calculations to confirm the validity of the code input data which describe the plant geometry and operating conditions, and also to confirm the Licensee's claim that the analysis of the BLDBF's was highly conservative. It was evident from the review that consistent, accurate data was being used and, furthermore, that there were procedures in place to minimise data errors. In addition, it was clear that conservative initial boundary conditions had been used, together with conservative assumptions for the core kinetic response, protection system response and the performance of the engineering safety features.

271. It was neither practicable nor necessary for NII to assess the transient analysis performed for every BLDBF against the plant limits. A judgement on the adequacy of the overall analysis could be formed by considering a selected number of BLDBF's. A significant level of effort was devoted to the assessment of the large LOCA BLDBF transient analysis, since historically this class of faults has been the subject of detailed scrutiny. The main concern has centred around the phenomenon known as clad ballooning. The methodology used for the analysis of large LOCA faults has developed significantly since the licensing of Sizewell B in 1987. The approach adopted by the Licensee of calculating the peak clad temperature (PCT) for comparison against the fuel rod plant safety limit temperature criterion takes account of the effect on PCT of clad ballooning and the uncertainties associated with code modelling and plant conditions. The combined analysis has calculated a PCT which is below the fuel rod plant safety limit temperature criterion by an acceptable margin. The small break LOCA BLDBF's were also subject to detailed assessment and, as a result, NII is satisfied the Licensee has demonstrated that adequate margins to the fuel rod structural integrity limit exist, without placing undue demands on the plant operator.

272. For intact circuit faults, a selection of BLDBF's which provided the most severe challenge to the plant limits were chosen for assessment. In each case, it was considered that the calculated margins to the plant safety limits were adequate. The NII review of the intact circuit fault BLDBF analysis has not revealed any issues which will undermine the Licensee's conclusion that the relevant plant safety limits are satisfied.

273. The NII assessment of the transient analysis work carried out against the radiological release plant safety limit was approached from two aspects:

- the validity of the radiological consequence analysis; and
- the comparison of the estimated consequences against the design targets.
274. Estimation of the radiological consequences from a given fault sequence was performed in three separate stages: estimation of the source term, estimation of the fraction of this released to the environment and estimation of the consequences of that release to the most exposed member of the public. Throughout this analysis, highly conservative data and assumptions have been used except where the results were considered to be close to the design targets, at which point an effort was made to remove highly conservative assumptions from the analysis to approach a more realistic prediction of consequences. NII gave consideration to the removal of these assumptions, and to the level of remaining conservatisms as part of the assessment of the adequacy of the total analysis.

275. The adequacy of the radiological analysis has been assessed by NII from consideration of the development of the radiological BLDBF's performed to estimate the frequency for each dose band, and of the subsequent radiological consequence calculations. Comparison of the radiological BLDBF's against the design targets has been considered by use of both the frequencies and consequences of the BLDBF's. Each radiological BLDBF has associated with it a radiological consequence and also an assigned frequency resultant from the summated frequencies of all the DBF's which it bounds. The Licensee concludes that the overall result of the analysis is conservative because in general the more frequent sequences tend to be bounded in radiological terms by less frequent but more onerous faults. The NII review of this analysis has not revealed any significant concerns. The radiological consequences arising from these fault sequences have been evaluated using a methodology which has been assessed, and is considered to be acceptable, by NII.

276. Comparison of the results with the design targets indicates that the summated frequencies in three of the consequence bands exceed the target values by factors of between two and five. The Licensee claims adequate compliance with the design targets on the basis of the degree of conservatism inherent in the analysis. Initially, NII did not fully accept this argument, on the basis that the analysis did not demonstrate compliance, nor provide an adequate ALARP case. However, as a result of further assessment using additional information provided by the Licensee, NII is now satisfied that an extensive and rigorous analysis has been carried out. NII is also satisfied that the final results are based upon conservative estimates and confirm the conclusion that adequate compliance with the design targets has been achieved, and that the risks are as low as reasonably practicable.

Decommissioning

277. At the end of its operational lifetime, the Sizewell B station will be decommissioned. This will need to be carried out in compliance with the legislation extant at that time. The essential requirements will be to ensure that the generation of radioactive waste is kept to a minimum and that radiation doses to workers and to the general public are as low as reasonably practicable.

278. It is a condition of the current nuclear site licence that the Licensee must make adequate arrangements for decommissioning. Prior to commencing any decommissioning activities, the Licensee will need to submit a
Pre-Decommissioning Safety Report to justify the particular methods to be adopted and explain how statutory requirements will be met.

279. In the design and pre-operational stage, the Licensee has had to demonstrate the feasibility of decommissioning the station in a safe manner, show that proper consideration has been given at the station design stage to minimise radioactive arisings (thereby reducing the potential levels of radiation doses) and quantify, as far as possible, the likely radioactive inventory.

280. In accordance with the recommendations of the International Atomic Energy Agency, the Licensee proposes to follow a three stage process for decommissioning, namely:

- remove the fuel from the reactor;
- clear all plant and buildings, except those immediately surrounding the reactor;
- eventual clearance of all radioactive materials from the site, followed by levelling and landscaping.

281. The Licensee has produced an outline plan to demonstrate the feasibility of carrying out this decommissioning process, using currently available technology. Under this plan, decommissioning could be completed within thirty years of the final shutdown. Other options, with extended timescales, have also been identified. The actual approach to be adopted will be determined closer to the end of the station's operational life and will be covered by the Pre-Decommissioning Safety Report.

282. The outline proposals have been reviewed by NII and are judged to be sufficient at this time. The feasibility of decommissioning the station in a safe manner has been demonstrated. The Licensee has also adopted sound design measures to reduce radioactive arisings and exposure levels during the decommissioning process.

Summary

283. On the evidence of the assessment carried out, which has involved a considerable level of effort over a number of years, NII is satisfied that the Licensee has produced a sound and comprehensive safety case for Sizewell B. The post-licensing commitments (para 25) have been satisfactorily completed, with the agreed exception of two items relating to in-service inspection, for which the Licensee is developing final proposals. Overall, NII agrees with the conclusions of the safety case and accepts the Licensee has demonstrated that the risks from Sizewell B are as low as reasonably practicable.

284. In a number of areas, notably the probabilistic safety analysis and the in-service inspection programme for the plant, the Licensee is continuing with work to develop and refine the safety case. These areas will be kept under review by NII to ensure their timely and satisfactory completion.
REGULATORY INSPECTION ACTIVITIES

Introduction

285. Throughout the construction and commissioning stages of Sizewell B, NII has carried out a range of inspection activities to monitor and regulate nuclear and radiological safety aspects. These activities have included:

- regular inspections for compliance with the nuclear site licence;
- team inspections, targeted on particular topics;
- specific (reactive) inspections to investigate occurrences or actions of interest and significance to nuclear safety;
- monitoring overall progress with and the witnessing of selected construction and commissioning activities;
- witnessing of emergency arrangements demonstration exercises;
- joint inspections with other regulatory authorities involved with Sizewell B; and
- exercising the licensing function and taking regulatory action.

286. The primary purpose of the inspection activities was to provide confirmation that the Licensee was carrying out construction and commissioning in accordance with nuclear safety requirements and was complying with the conditions of the nuclear site licence. The specific nature and purpose of the principal inspection activities (above) are described in the following sections. The inspections have been carried out by the nominated NII inspectors for the Sizewell B site supported, as necessary, by other colleagues including specialist inspectors from the two assessment branches of NII (see para 20).

Regular (Basic) Inspections

287. The nominated NII site inspector makes regular visits to the site to examine the Licensee’s arrangements for ensuring safety and to inspect for compliance with the conditions attached to the nuclear site licence. With the introduction in 1990 of a standard licence, there are thirty-five conditions (see Table 7). These conditions define the areas to which a licensee should pay particular attention to safety matters. In the main, the conditions require a Licensee to make and implement adequate arrangements to address the particular aspects identified.

288. Although the conditions are standard, each Licensee can develop and tailor arrangements to suit. The arrangements for compliance are likely to change as a licensed site evolves, from initial design and construction through operation to final
decommissioning. This has been the case for Sizewell B through the construction and commissioning stages.

289. Inspection for licence compliance is carried out to a basic inspection programme (BIP). This programme ensures that all licence conditions are inspected on a regular basis and places the emphasis (in terms of the inspector’s time) in accordance with the relative significance to safety and the nature of the condition. For example, more time is afforded to inspect for compliance with the condition relating to operating rules (Licence Condition 23) than for marking of the site boundary (Licence Condition 2). This approach to regular inspection is applied by NII at all sites subject to the standard nuclear site licence; it is not unique to Sizewell B.

290. In considering the arrangements for licence compliance for Sizewell B, NII sought consistency with arrangements, practices and procedures already in place at the Licensee’s other licensed sites (gas-cooled reactor stations). Exceptions to this general principle could be made where the nature of the plant (a pressurised water reactor) demanded a different approach and/or where the proposed approach had obvious advantages.

291. One area where the arrangements, practices and procedures differ markedly for Sizewell B, compared with the gas-cooled stations, is in the use of technical specifications as the statement of operating conditions and limits. The technical specifications define, in a structured format, the safe operating regimes for the reactor plant. They also stipulate the surveillance (inspection and test) requirements to demonstrate plant operability.

292. The use of technical specifications is common practice world-wide on pressurised water reactor plants, most notably in the United States. The technical specifications for Sizewell B form an integral part of the Licensee’s arrangements for licence compliance (specifically for Licence Conditions 23, 24 and 28 - refer to Table 7). NII has carefully considered these arrangements and the acceptability of the use of technical specifications within the regulatory framework in the United Kingdom. This consideration included a targeted (team) inspection to examine the acceptability of the technical specifications for Sizewell B - this team inspection is discussed in the following section (page 61). NII also employed consultants to review world-wide practice in the use of technical specifications. In conclusion, NII accepted the use of technical specifications and their application within the arrangements for licence compliance.

293. The Licensee’s arrangements (under Licence Condition 10) for the training of staff have been developed specifically to suit the requirements of a pressurised water reactor. These encompassed training of plant operators in the use of technical specifications. The Licensee employed a full-scope control room simulator to facilitate training under realistic conditions in accordance with a requirement made by NII in 1987 (see para 13). NII has reviewed the training strategy and arrangements, including the use of the simulator. Progress on the training of personnel has been regularly inspected. The training strategy and the arrangements were and are judged to be acceptable.
294. A prerequisite for the granting of the licence to construct Sizewell B in 1987 (see para 12) was the Licensee should have in place a suitable process for controlling post-licensing design changes. When the standard licence was introduced in 1990, this process was encompassed in the arrangements for compliance with Licence Condition 20 - modification to design of plant under construction. In accordance with the licence condition, the arrangements provide for classification of modifications in terms of safety significance. The Licensee appointed a safety design change committee (SDCC) to review proposals for modifications, confirm the safety classification and approve (or reject) the modification. The arrangements also required that those proposals with the highest safety classification were forwarded to NII for agreement, subsequent to approval by the SDCC. NII is satisfied that the Licensee's arrangements were and are sound with respect to control of modifications. There was one occasion during construction when the arrangements were found to have been infringed; NII took regulatory action for this breach of the licence condition (this is explained further in the section on Exercising the Licensing Function, para 350).

295. Other points of note, where the arrangements were developed specifically for Sizewell B, relate to the nuclear safety committee (Licence Condition 13) and to commissioning (Licence Condition 21). The Licensee established a nuclear safety committee dedicated to pressurised water reactors, when the standard licence was introduced in 1990. The committee, which includes members who are independent of the Licensee, considers and advises the Licensee on matters of nuclear safety. It is the first time a nuclear safety committee has been formed for a nuclear power station under construction in the United Kingdom. The terms of reference of the committee have been approved by NII.

296. The arrangements for commissioning of Sizewell B needed to be developed to take into account the pressurised water reactor and its ancillary plant. NII carried out a team inspection in 1992 to review the arrangements for commissioning (further details are given in the following section, page 61). NII also carried out a sample review of the experience and training of personnel involved in commissioning activities. This established there was a significant background of experience with pressurised water reactors amongst the personnel involved. NII is satisfied that the arrangements provided for appropriate control over the nuclear safety related aspects of the commissioning programme.

Team Inspections

297. Team inspections comprise a number of NII inspectors working together in a systematic and co-ordinated manner to closely examine a selected topic. Such inspections are more intensive, and enable a more detailed appraisal to be carried out, than is practicable during the regular (basic) inspection programme. Typically, a team inspection comprises four to eight inspectors and lasts from three to five days. Team inspections are used selectively by NII and are targeted on areas or activities of particular interest or significance.

298. For Sizewell B, team inspections have been carried out to examine topics such as:
• the arrangements for commissioning;
• the acceptability of technical specifications;
• the radioactive waste facility;
• readiness for loading fuel into the reactor;
• readiness for operation;
• the operational safety case; and
• completion of commissioning.

The Arrangements for Commissioning

299. In 1991, when granting consent for stage 1 of the commissioning phase (see para 18), NII advised the Licensee that the arrangements for the later stages would necessarily need to be more comprehensive. To confirm that the Licensee had acted accordingly, and had adequate arrangements for these later stages, NII carried out a team inspection in 1992. This was a particularly extensive inspection and involved a team of eight, including two members from the US Nuclear Regulatory Commission, over a period of three weeks.

300. The specific objective of the inspection was to confirm the Licensee had in place comprehensive and transparent arrangements, procedures and controls for the commissioning of Sizewell B. The areas examined by the team included:

• the organisation and management of commissioning;
• the conduct of commissioning operations;
• the interfaces between commissioning and other activities; and
• the discharge of safety case commitments during commissioning;

301. The overall conclusion from the inspection was that the arrangements were satisfactory and were compliant with Licence Condition 21. As usual on such inspections, there were a number of findings and recommendations to be addressed by the Licensee. These were subsequently closed out to the satisfaction of NII.

The Acceptability of Technical Specifications

302. The introduction of technical specifications was new to the United Kingdom (para 291). NII had agreed in principle to their use for Sizewell B, subject to confirmation that they were acceptable and were enforceable within the UK licensing regime. To seek this confirmation, NII carried out a team inspection in 1993. Again, the team included members from the US Nuclear Regulatory Commission.
303. There were three broad objectives for the inspection, namely:

- to determine whether the technical specifications were likely to form an acceptable means of complying with the relevant licence conditions;

- to confirm the technical specifications were an adequate representation of the safety case (and the appropriate conditions and limits therein were reflected in the technical specifications); and

- to audit the process of producing the technical specifications.

304. In fulfilling these objectives, one of the approaches employed was to carry out a sample comparison between established operating rules for gas-cooled reactors and the equivalent technical specifications. The team concluded that the technical specifications compared favourably and their simplicity and "user friendliness" were commended.

305. To form a view on the legal enforceability of technical specifications (with respect to licence compliance), the team used case studies representing possible events which could result in regulatory action. The case studies were set to test whether there was common understanding and interpretation between regulator and operator in applying the technical specifications. These studies proved successful; there were some matters arising relating to enforceability which NII raised and resolved with the legal department of the Health and Safety Executive.

306. There were a number of findings and recommendations arising from the inspection to be addressed by the Licensee. In particular, the requirements of Licence Condition 28 regarding examination, inspection, maintenance and testing of plant needed to be reconciled with the surveillances stipulated in the technical specifications. The findings and recommendations have been satisfactorily closed out. NII accepts that the Sizewell B technical specifications are a satisfactory means of achieving compliance with the relevant nuclear site licence conditions.

The Radioactive Waste Treatment Facility

307. The radioactive waste treatment facility at Sizewell B is an important installation (as described in the section on Radioactive Waste Management, page 40). In advance of the facility becoming operational, NII carried out a team inspection in 1994. The inspection was timed to allow ample time for any findings to be resolved before the facility would be required to receive radioactive waste, from the time the reactor first achieved criticality.

308. The objective of the inspection was to determine if there were any regulatory obstacles to the plant being committed into operation in support of the reactor at the time of first (initial) criticality. Particular areas targeted during the inspection were the handling and storage of primary and secondary waste resins, and the gaseous waste system. Selected plant commissioning and operating procedures were reviewed. The findings arising from the inspection, many of which were related to
finalisation or clarification of operating procedures, were closed out to NII's satisfaction, as a prerequisite to agreement for initial criticality.

Readiness for Fuel Load

309. The loading of nuclear fuel into the reactor was a significant stage in the commissioning programme for Sizewell B. It marked the start of the period in which the reactor would be prepared for operation and be taken in stages from initial criticality up to full power. NII carried out a team inspection on readiness for fuel load in 1994, shortly before the Licensee's planned date for this event. The NII team included a member from the Swedish nuclear inspectorate (SKI).

310. The objective of the inspection was to examine the Licensee's arrangements and preparedness for fuel load. This stage involved far more than the activity of loading fuel into the reactor. Commissioning of certain reactor systems had to have been successfully completed, the operating personnel had to be ready to enter this key stage and needed to apply the appropriate technical specifications. To gain confidence that an acceptable state of readiness had been achieved, the team inspection examined a number of facets, including:

- the competence of operational teams and their familiarity with the requirements of the relevant technical specifications;
- the process for and the control over taking plant systems into service;
- the controls in place to ensure the plant operators were presented with accurate information on plant status; and
- inspection of a sample of plant systems to confirm their readiness and the standard of completion.

311. The inspection team concluded that high standards were being achieved in systems engineering, a positive attitude to safety was evident and, overall, the licensee had a robust process in place to demonstrate readiness for fuel load. The findings from the inspection were satisfactorily closed out prior to the granting by NII of consent to load fuel.

Readiness for Operation

312. Prior to initial criticality, and as a follow-on to the fuel load readiness inspection, NII carried out a team inspection on operational readiness at the end of 1994. The primary objectives were to examine compliance with the technical specifications and the state of plant readiness.

313. The inspection was timed to coincide with the point in commissioning where a significantly greater proportion of the technical specifications became applicable and where most of the plant not required at fuel load stage needed to be available. Three areas were examined, namely:
- familiarity and compliance with technical specifications (including arrangements for complying with operating conditions and limits);
- the knowledge from and the use made of experience (lessons learned) from initial use of technical specifications; and
- plant readiness.

314. The team found a number of incidences where compliance with the technical specifications had not been achieved. These generally stemmed from the fact that operators were still gaining experience and needed to clarify or refine practices and procedures, rather than from any fundamental problem. The Licensee had already taken initiatives to address the errors in compliance, including review of the content and presentation of the technical specifications and the responsibilities for achieving and monitoring compliance. The inspection team concluded that there were no significant obstacles to NII agreement to initial criticality. NII will continue to monitor compliance with, and operational experience from, the use of technical specifications at Sizewell B.

The Operational Safety Case

315. The safety case for Sizewell B consists of a large number of documents, of different types and purpose. These have been produced and reviewed by the Licensee, in accordance with defined arrangements, during the design, construction and commissioning stages. It is important for the station, and from a regulatory perspective, for the safety case documentation to be clearly delineated and maintained throughout life.

316. To ensure that the suite of documentation which comprises the safety case is both clearly defined and is readily accessible for operational use, the Licensee is producing a set of user guides. These explain the basis of the safety case and identify relevant documentation, with each guide covering a defined system or topic area. The complete set of user guides, in conjunction with the Station Safety Report, will be held by Sizewell B station. They will be utilised should the safety case need to be reviewed or revised in the future (for example, in consideration of a proposed modification).

317. NII carried out a team inspection in 1995 to examine the process for production of the user guides. The primary objectives of the inspection were to establish confidence that the process, as set down by the Licensee, was being applied correctly and to consider the adequacy of the output from this process. To do this, the team looked at selected areas, covering different aspects of the safety case.

318. The team concluded that the process was sound and had been applied in a thorough and systematic manner. The user guides were judged to provide a workable route map to, and around, the safety case documentation. There were a number of detailed findings which were closed out to the satisfaction of NII. The licensee is continuing with the production of user guides, to a programme agreed

64
with NII. It is intended that NII will carry out a further review of the user guides when the work has been completed.

Completion of Commissioning

319. In advance of granting consent to commence full operational use of Sizewell B, NII carried out a team inspection on the completion of commissioning. The objectives of the inspection were to confirm:

- the commissioning activities had reached a satisfactory state of completion;
- the necessary changes to station documentation had been identified and implemented (or were in process);
- the necessary changes to licence compliance documentation had been prepared; and
- the material condition of the plant was satisfactory.

320. The inspection team found both plant and procedural aspects to be in a satisfactory state of completion. The necessary revisions to the arrangements for compliance with the conditions of the nuclear site licence were also in a satisfactory state of preparation. The licensee's provisions for the control and completion of outstanding work were examined and judged to be acceptable.

Specific (Reactive) Inspections

321. Matters can come to the attention of the NII site inspector which warrant specific investigation. These can arise whilst on site during a basic or team inspection (as described in the previous sections). Alternatively, they can be identified by specialist NII inspectors or other site inspectors during the course of their work. The Licensee can also notify the site inspector of specific events or activities. Following-up such matters falls into the category of reactive inspection.

322. Reactive inspections, by their nature, encompass a wide range of activities. Typical examples are the investigation of incidents, the inspection of particular facilities or items of plant and the review of working practices or procedures. The NII site inspector can, if required, enlist the help of specialist inspectors to assist with the task.

323. Some of the more notable reactive inspections on Sizewell B are described below. In all cases, the inspection has been successfully completed and the matter concluded to the satisfaction of NII.

324. During the construction phase, the reactor pressure vessel was delivered to and stored on the site, prior to installation in the reactor building. The site inspector, with advice from specialist inspectors, obtained information from the Licensee regarding the on-site storage facilities and the procedures for care of the vessel.
The provisions for storage and care were inspected and, after NE had made some improvements, were judged to be acceptable.

325. An unusual incident occurred in 1990, when an intruder to the construction site was apprehended. The intruder claimed to have placed salt in various areas of concrete around the site. NII monitored the Licensee's investigations to establish whether or not there was any substance to the claim. Tests carried out by the Licensee showed no evidence of salt having been placed in areas of (at the time) moist concrete.

326. In 1993, whilst a routine commissioning check was being performed on a reactor coolant pump (which involved turning the pump shaft by hand), it was found that the pump would not rotate. The problem was traced to the shaft seal package. The Licensee informed the site inspector, who was present to witness the dismantling of the seal assembly. The inspector subsequently progressed with the Licensee a thorough investigation into the reason for the problem and checks to ensure that none of the other three reactor pumps were similarly affected. The investigation concluded that the seal had been incorrectly installed. Steps were introduced by the pump manufacturer to prevent a recurrence of the problem. The other pumps were found to be unaffected. The matter was cleared by NII prior to the granting of consent to load fuel.

327. Testing during Stage 1 of commissioning (see para 18) revealed that the operation of the emergency boration system (EBS) isolating valves, between the borated water tanks and the reactor primary circuit, gave rise to severe pressure transients within the tanks. The Licensee brought the problem to the attention of NII and was able to show that the effect was due to the speed of opening of the valves which, in combination with the pressure difference across them, caused an effect similar to water hammer. A re-analysis of the system performance was carried out. This established that a slower valve opening time would preclude the problem and would meet the safety case requirement for the system. The supporting analysis was discussed with NII prior to changes being implemented on the plant. Testing of the revised installation was carried out prior to the start of Stage 2 commissioning. A further confirmatory test was successfully completed during this stage.

328. During Stage 2 commissioning, a fracture occurred in a welded connection between a small bore drain line and its isolating valve on one of the steam generators. Investigations revealed that this was a stress induced fracture caused by vibration in the line. The vibration was confined to this particular section of drain line and occurred only on one steam generator. The cause of this vibration was determined to have been vortexing of liquid flow through a drain hole within the steam generator lower head baffle plate, which set up a vibration within the drain line. Repairs to the drain line were witnessed by NII. Subsequently a revised drain line design, the results from examination of the other baffles and work to support the causal analysis were reviewed and cleared by NII prior to the granting of consent to load fuel.

329. After completion of the hot functional test in Stage 2 commissioning, the Licensee reported to NII that attempts had proved unsuccessful to remove two of the
fifty four studs which are used to secure the closure head of the reactor pressure vessel in position. The studs were being taken out so that the closure head could be removed, in preparation for the loading of fuel into the reactor vessel. Following discussions between the Licensee and NII on an appropriate method of repair, the two studs were cut off and removed. All studs and stud holes were examined for signs of thread damage which could give rise to future problems and minor repairs (within the original design tolerances) were carried out to several studs and holes. The necessary remedial work was completed to the satisfaction of NII before fuel load.

330. During a routine visit to site in 1994, the site inspector was informed of a concern regarding sulphur contamination of the contact faces on one particular make of electrical relay. The inspector progressed the reporting of the investigation into the reasons for and implications of the contamination. It was established by the Licensee that sulphur was present in a rubber sealing ring used in the relay. This sulphur could result in a build-up of silver sulphide fur on the silver-plated contact faces of the relay. The seals were not required for the safe operation of the relays and could be discarded pending the development of a more suitable sealing material. All contact faces were checked and, where necessary, cleaned. This matter was resolved to NII's satisfaction prior to the granting of consent to load fuel. Subsequently, further inspections by the licensee have been carried out to confirm that the build-up of fur has not re-occurred.

331. In the course of the NII team inspection on readiness for fuel load, some of the team members raised queries regarding the alarm display system in the main control room. The queries were not directly relevant to the objectives of the team inspection and were therefore raised separately by the site inspector, by writing to the Licensee. With the assistance of a specialist inspector, these queries were progressed by the site inspector. The licensee proposed and has implemented an optimisation scheme for the alarm system. Under this scheme, a number of enhancements to the system are being considered to assist the plant operators to process alarms. The work is planned to be completed within the first cycle of operation.

Monitoring Progress Of Construction And Commissioning

332. The NII site inspector was responsible for monitoring overall progress of construction and commissioning. In particular, the site inspector maintained an overview of the Licensee's progress towards completion of the activities linked to each of the licensing hold points in the construction and commissioning programme (as described in para 15).

333. The site inspector could identify when key activities were likely to take place. This information could then be used to arrange for NII involvement in the witnessing of specific events or tests. In some cases the involvement was restricted to the site inspector, in others it included specialist inspectors, depending upon the nature of the activity.
334. Prior to witnessing an activity the inspector(s) involved would usually review at least a sample of the relevant documentation (eg principles, procedures) produced by the Licensee. The act of witnessing generally encompassed a number of aspects, such as:

- the management and implementation of the activity or test;
- familiarity with and adherence to relevant procedures and quality standards;
- awareness of nuclear safety implications;
- recording and analysis of events or data; and
- demonstration of successful completion.

335. In the construction phase (see para 16), the formal stages witnessed by NII included installation of the reactor pressure vessel and the primary circuit hydrostatic test. For a later hold point, namely delivery of fuel to the site, NII reviewed the measures put in place by the Licensee to prevent inadvertent criticality or damage to the fuel and inspected the new fuel storage area prior to granting consent to begin delivery of the fuel. NII then witnessed the handling of the initial delivery.

336. Within each formal stage of the commissioning programme (see para 18) there were numerous individual tests to be carried out. At the start of each stage the Licensee compiled a schedule, known as the safety inspectorate schedule (SIS), of those tests which were considered to be of particular significance to nuclear safety. NII would then nominate tests from the SIS which were to be witnessed. The nominated tests could not be carried out without the Licensee first giving notification of when they were to take place and inviting NII to be present.

337. The nominations for witnessing of tests on the SIS were compiled by the NII site inspector, based upon advice from specialist inspectors. The Licensee needed to define the requirements and the success criteria in advance of each test, or series of tests. Witnessing of tests would normally involve the appropriate specialist inspectors in addition to, or in place of, the site inspector. All completion certificates for tests on the schedule had to be submitted to NII.

338. Given that Sizewell B was the first of its type in the UK, NII took a particular interest in commissioning tests which were fundamental to the design concept of the pressurised water reactor. Two examples were the structural over-pressure test of the primary containment building and reactor physics tests. The over-pressure test demonstrated that the containment structure could withstand the internal pressure generated under postulated reactor accident conditions. Reactor physics tests demonstrated that the reactor core behaves as designed and predicted.

339. In the case of the reactor physics tests, NII first reviewed and agreed the paper of principle. In doing so, NII consulted with the US Nuclear Regulatory
Commission (NRC) in light of their wide experience with pressurised water reactors. A member of the NRC with expertise in pressurised water reactor physics was present, to provide assistance to NII, during witnessing of the approach to initial criticality at Sizewell B and for the low power physics testing.

340. During the later stages of commissioning, many integrated plant tests were being carried out. To enable NII to better organise attendance at specific tests, the Licensee submitted more frequent updates of the programme of tests. NII also increased the presence on site of inspectors, by the formation of a group led by the nominated site inspector.

Witnessing Of Emergency Exercises

341. It is a condition of the nuclear site licence that the Licensee shall make and implement adequate arrangements for dealing with any accident or emergency arising on the site and their effects. It is also a requirement that the arrangements are rehearsed at regular intervals. The rehearsals are carried out by way of emergency exercises. Each exercise for the purpose of demonstrating the adequacy of the arrangements is based upon an accident scenario agreed with NII.

342. Throughout the construction and commissioning stages at Sizewell B, emergency exercises have taken place at regular intervals. All the formal demonstration exercises have been witnessed by NII. The scenarios for the exercises have evolved to reflect the status of the reactor plant and the nature of the potential hazard. In the earlier stages, the exercises reflected the fact that the arrangements needed to address the potential hazard from the adjacent Sizewell A station. For the later stages (from when nuclear fuel was first present on site), the emergency arrangements and the exercises have been developed to take into account the potential nuclear hazard from Sizewell B itself. The full-scope control room simulator has been used to good effect to run the accident scenarios in these later stages.

343. Important stages in the development and demonstration of the Licensee’s emergency arrangements were linked to the hold points for the delivery of new fuel to site and the loading of fuel into the reactor. Completion of a successful demonstration exercise was a prerequisite for NII granting consent to progress beyond each of these hold points.

344. The emergency arrangements now in place at Sizewell B are based upon arrangements in place at all the Licensee’s nuclear sites. The most recent successful demonstration of the arrangements took place at Sizewell B in May 1995. Another emergency exercise will take place in 1996.

Joint Inspections

345. As described in the section on the Regulatory Process (para 8), there are other bodies which have regulatory enforcement and inspection responsibilities at Sizewell B. These are the Field Operations Division (FOD) of the Health and Safety
Executive, Her Majesty's Inspectorate of Pollution (HMIP) and the Ministry of Agriculture, Fisheries and Food (MAFF).

346. Liaison arrangements have been established between NII and these other bodies to ensure effective communications and co-ordination of activities. For Sizewell B, a number of joint inspections have taken place between the nominated NII site inspector and (at different times) counterparts from FOD, HMIP or MAFF in areas of common interest. For example, joint inspections of the radioactive waste facility have involved NII and HMIP.

Exercising The Licensing Function

347. The section on the Regulatory Process (page 1) explains the licensing function, as applied to Sizewell B, and the regulatory powers available to NII. It is primarily the responsibility of the power reactor inspection Branch (see para 20) to carry out the necessary regulatory actions within this process.

348. Regulatory control over the construction and commissioning stages has been effected by use of the powers available to NII under the nuclear site licence and the Health and Safety at Work Act (see para 7). Predominantly, this has involved the granting of consents for the Licensee to progress beyond each of a series of defined hold points (see Table 2).

349. In addition, a number of specifications were issued when NII required the Licensee to implement certain arrangements. One example is a specification issued in 1992 which required the Licensee to have the consent of NII prior to starting any of the commissioning stages (see para 19). Another specification (issued in 1993) required that all design changes to the software for the primary protection system be referred to NII for agreement prior to their implementation. This amended and was additional to the Licensee's then current arrangements for design changes (modifications) which required only those of the highest safety classification to be forwarded to NII (see para 294). The specification was subsequently withdrawn in 1994, once NII had been satisfied that the Licensee's arrangements for the classification and review of proposed software changes were acceptable.

350. Regulatory enforcement action was judged to be necessary on one occasion. The NII site inspector discovered that the Licensee had not complied with the arrangements for control of modifications (see para 294) in that work to implement a modification on a reactor auxiliary system had been started before the Licensee had completed the necessary processes which included the need to obtain NII agreement. Firstly, NII issued a direction to halt the modification work in question until the matter had been resolved. It was established there were differences between the Licensee's arrangements and the procedures for modifications. To preclude a recurrence of the situation, NII issued an improvement notice which required the Licensee to review and revise the working procedures to ensure the appropriate arrangements were implemented. The improvements were effected and the notice discharged within one month. The modification itself was judged to be acceptable by NII and was ultimately implemented successfully.
Other Activities

351. Outwith the regulatory aspects, NII (in particular the site inspector) has had an involvement with a number of other activities relating to Sizewell B. There have been many visits to the site by members of nuclear regulatory authorities in other countries, such as Russia, the United States and the Netherlands. There was also a visit by an international working group on inspection practices. In most cases, in addition to gaining information about Sizewell B, the visitors have discussed with NII the regulatory framework and the activities involved in licensing. In similar vein, a member of the Swedish nuclear inspectorate (SKI) was attached to NII for six months in 1994 and worked predominantly with the Sizewell B Site Inspector. Members of the Health and Safety Commission (HSC) and of the HSC’s Advisory Committee on the Safety of Nuclear Installations (ACSNI) have also visited Sizewell B, in the company of NII.

352. In 1992, at the request of the UK Government, the International Atomic Energy Agency sent a team of international experts in the nuclear field to Sizewell B to review commissioning activities and the preparations for operation. This pre-operational safety review team (Pre-OSART) provided the opportunity for an exchange of knowledge and experience between the team members and their counterparts within the Licensee’s organisation on how the goal of excellence in operational safety could be further pursued. The work of NII was not within the remit of the review, though aspects of the regulatory interface between the Licensee and NII were examined. During the review, the NII site inspector was available to provide explanation and assistance to the team. A report on the findings from the review was submitted to the Government by the International Atomic Energy Agency in 1993. Progress with the Pre-OSART recommendations was confirmed during a revisit by some of the team to Sizewell B in February 1994. Prior to granting consent to load fuel in September 1994, NII confirmed that the outstanding findings from the review had been satisfactorily closed out by the Licensee.

353. The NII site inspector regularly attends meetings of the Sizewell local community liaison committee. In addition, a quarterly report on NII’s inspection activities is submitted to the committee. This practice is common to all licensed sites.

Summary

354. NII is satisfied that the Licensee has demonstrated compliance with the conditions of the nuclear site licence and that the construction and commissioning stages at Sizewell B have been carried out with due regard to nuclear safety. This conclusion is a result of the regular and wide ranging regulatory inspection activities carried out by NII during these stages.

355. Throughout the lifetime of Sizewell B nuclear power station, NII will continue to carry out regulatory inspections. These will comprise regular (basic) inspections, augmented as necessary by other appropriate activities.
COMPLETION OF COMMISSIONING

356. The Licensee has completed all the defined stages of commissioning. Progression to each successive stage has required the consent of NII, as described in the section on The Regulatory Process (paras 15, 18 and 19). The Licensee's arrangements for commissioning were examined by NII and found to be acceptable, as described in the section on Team Inspections (paras 299 to 301).

357. Commissioning has generally proceeded as planned by the licensee. The majority of problems which were encountered related to the conventional items of plant and equipment, rather than the reactor systems. The more notable occurrences on the latter are described in the section on Specific Inspections (paras 326 to 330).

358. The last stage of commissioning was the final power raise period, when the reactor was taken progressively from 60% to 100% power. At the end of this period, a successful demonstration of safe reactor shutdown (trip) from full power took place. This was followed by a natural circulation test, which demonstrated that adequate cooling of the reactor can be achieved with no reactor coolant pumps running.

359. The completion of commissioning marks the transition point for commencement of normal, commercial operation of Sizewell B power station. Under the conditions of the nuclear site licence (specifically, Licence Condition 21 - see Table 7) NII consent was required for the licensee to commence operating the station for purposes other than commissioning. Before granting consent, NII needed to be satisfied the licensee was ready to make the transition to operational mode.

360. NII required the licensee to submit a formal statement of readiness for normal (commercial) operation. This encompassed the readiness of plant, procedures and personnel. A meeting took place between NII and the licensee to discuss and confirm the prerequisites for the granting of consent (as is customary in advance of an application for consent). NII carried out a team inspection on the completion of commissioning, to confirm a satisfactory state of readiness (see page 65).

361. In accordance with the established liaison agreements (see para 8), NII also consulted with the other regulatory bodies involved at Sizewell B. These are the Field Operations Division (FOD) of the Health and Safety Executive, Her Majesty's Inspectorate of Pollution (HMIP) and the Ministry of Agriculture, Fisheries and Food (MAFF). This established there were no other regulatory matters outstanding which would affect the granting of consent by NII to commence normal operation of the station.

362. NII is satisfied that commissioning of Sizewell B is complete and has achieved the required objectives. NII is also satisfied that the licensee has demonstrated readiness to make the transition from the commissioning phase to operational use of the station and has met the prerequisites set down by NII.
PLANNED DÉVELOPMENTS

363. The Licensee has notified NII of planned developments to the design and operation of Sizewell B. These are intended to improve the operability, availability and performance of the station.

364. The first of the proposed changes is the introduction of automatic frequency responsive operation (AFRO). This will be additional to the normal base load mode of operation. In frequency responsive mode, the station will respond to changes in frequency on the national grid; this will help to stabilise the grid system. The Licensee has contracted to provide frequency responsive operation to the national grid, for a limited time within the first cycle of operation (up to the first reactor refuelling outage), subject to agreement by NII. A safety case to justify this mode of operation for cycle 1 has been submitted to NII and is judged to be satisfactory. The licensee needs to complete the necessary amendments to operating procedures, provide additional operator training and carry out commissioning tests before the AFRO mode is utilised.

365. For the second and subsequent cycles of operation, the Licensee is proposing a series of changes which encompass further frequency responsive operation, the use of more advanced designs for reactor fuel and core components, and improvements in fuel cycle efficiency. A revised safety case, covering the effects of these changes, will need to be produced by the Licensee. The changes will not be implemented without the agreement of NII.

CONCLUSIONS

366. NII, on behalf of the Health and Safety Executive, has granted consent (on 22 September 1995) for the Licensee to take Sizewell B nuclear power station into full operational use. The consent has been granted on the basis that the Licensee has satisfactorily completed all construction stages and all commissioning stages defined under the licensing process. A satisfactory safety case has been produced and the Licensee has adequately demonstrated compliance with the conditions attached to the nuclear site licence.

367. While the granting of consent to commence full operational use is a significant milestone, it does not constitute the completion of regulatory activities for Sizewell B. NII will continue to regulate activities at Sizewell B throughout the lifetime and eventual decommissioning of the station.

368. The Licensee has revealed plans to introduce developments to the design and operation of Sizewell B. The changes will require the agreement of NII before they can be implemented.
REFERENCES


### TABLE 1 - MILESTONES IN THE SIZEWELL B LICENSING PROGRAMME

<table>
<thead>
<tr>
<th>MILESTONE</th>
<th>DATE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Application for Section 2 Consent¹</td>
<td>January 1981</td>
</tr>
<tr>
<td>Application for Nuclear Site Licence²</td>
<td>January 1981</td>
</tr>
<tr>
<td>Draft PCSR Submitted to NII</td>
<td>December 1981</td>
</tr>
<tr>
<td>Public Inquiry (chaired by Sir Frank Layfield)</td>
<td>January 1983 to March 1985</td>
</tr>
<tr>
<td>Public Inquiry Report Published</td>
<td>January 1987</td>
</tr>
<tr>
<td>Section 2 Consent Granted¹</td>
<td>March 1987</td>
</tr>
<tr>
<td>Final Issue of PCSR</td>
<td>May 1987</td>
</tr>
<tr>
<td>Revised Nuclear Site Licence Issued²</td>
<td>June 1987</td>
</tr>
</tbody>
</table>

**Notes:**

1. Consent for the construction of the power station was required under Section 2 of the Electric Lighting Act 1909. This was granted by the Secretary of State for Energy.

2. A nuclear site licence was already in existence for the Sizewell site, covering Sizewell A Megnox station. The site licence was revised to encompass Sizewell B station.
### TABLE 2 - LICENSING STAGES IN CONSTRUCTION AND COMMISSIONING

<table>
<thead>
<tr>
<th>LICENCE STAGE</th>
<th>DESCRIPTION</th>
<th>CONSENT&lt;sup&gt;1&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>No.</td>
</tr>
<tr>
<td>Construction Stages</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Stage 1</td>
<td>Pouring of foundation mass concrete</td>
<td>3</td>
</tr>
<tr>
<td>Stage 2/1</td>
<td>First permanent structural concrete</td>
<td>6</td>
</tr>
<tr>
<td>Stage 2/2</td>
<td>(carried out in sub-stages 1 to 4)</td>
<td>8</td>
</tr>
<tr>
<td>Stage 2/3</td>
<td></td>
<td>10</td>
</tr>
<tr>
<td>Stage 2/4</td>
<td></td>
<td>14</td>
</tr>
<tr>
<td>Stage 3</td>
<td>Mechanical access to radwaste building</td>
<td>14</td>
</tr>
<tr>
<td>Stage 4</td>
<td>Installation of reactor pressure vessel</td>
<td>17</td>
</tr>
<tr>
<td>Stage 5</td>
<td>Installation of reactor coolant pump support legs</td>
<td>18</td>
</tr>
<tr>
<td>Stage 6</td>
<td>Commence pre-stressing of primary containment</td>
<td>22</td>
</tr>
<tr>
<td>Stage 7</td>
<td>Commence full-scope simulator training</td>
<td>26</td>
</tr>
<tr>
<td>Stage 8</td>
<td>Commence testing of primary protection system</td>
<td>29</td>
</tr>
<tr>
<td>Stage 9</td>
<td>Primary circuit hydrostatic test</td>
<td>35</td>
</tr>
<tr>
<td>Stage 10</td>
<td>Fuel load</td>
<td>49</td>
</tr>
<tr>
<td>Commissioning Stages</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Stage 1</td>
<td>Phase 1 - Commencement of initial plant testing</td>
<td>19</td>
</tr>
<tr>
<td></td>
<td>Phase 2 - Completion of plant functional testing</td>
<td>25</td>
</tr>
<tr>
<td>Stage 2</td>
<td>System and, integrated test period (HFT)</td>
<td>46</td>
</tr>
<tr>
<td>Stage 3</td>
<td>Fuel load and low power testing up to 5% power&lt;sup&gt;2&lt;/sup&gt;</td>
<td>48</td>
</tr>
<tr>
<td>Stage 4</td>
<td>Initial raise power period from 5% to 60% power</td>
<td>52</td>
</tr>
<tr>
<td>Stage 5</td>
<td>Final raise power period from 60% to full power&lt;sup&gt;3&lt;/sup&gt;</td>
<td>53</td>
</tr>
</tbody>
</table>

Notes:

1. Consents for construction stages 1 and 2 were granted under version 12B of the Sizewell nuclear site licence. Consents for stages 3 to 10, and for the commissioning stages, were granted under licence version 12C. This is why there are two consents with the same number (14).

2. After Stage 3 consent, the Licensee had to obtain NII agreement to take the reactor critical and commence power raise.

3. After Stage 5 consent, a further consent from NII was required to take the station into full commercial operation. This latter consent (no. 56) was granted on 22 September 1995.
## TABLE 3 - NII POWERS UNDER A NUCLEAR SITE LICENCE

<table>
<thead>
<tr>
<th>POWER</th>
<th>EXPLANATION</th>
<th>REASON FOR USE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Consent</td>
<td>A consent is required before a licensee can carry out any activity for which NII has so specified the need.</td>
<td>A consent is used to ensure a licensee does not carry out an activity before NII has been satisfied that the proposed course of action is safe and all necessary procedures and controls are in place.</td>
</tr>
<tr>
<td>Approval</td>
<td>A licensee is required to submit its arrangements for approval, if so specified by NII.</td>
<td>An approval is used to freeze a licensee's arrangements. Once arrangements are approved, no alteration or amendment can be carried out without the further approval of NII.</td>
</tr>
<tr>
<td>Direction</td>
<td>A direction requires a licensee to take a particular action.</td>
<td>A direction is used for matters of major or immediate safety importance.</td>
</tr>
<tr>
<td>Agreement</td>
<td>An agreement allows a licensee to proceed, in accordance with its own arrangements.</td>
<td>Where the need to obtain NII agreement is written into the licensee's arrangements, it prevents a licensee from proceeding unless the course of action has been agreed.</td>
</tr>
<tr>
<td>Notification</td>
<td>When so notified, a licensee is required to submit information to NII.</td>
<td>A notification to a licensee is used to request the submission of information to NII.</td>
</tr>
<tr>
<td>Specification</td>
<td>As specified by NII, a licensee is required to implement the specified arrangements.</td>
<td>A specification is the means by which NII can implement discretionary control over a licensee's arrangements.</td>
</tr>
</tbody>
</table>
TABLE 4 - ALLOCATION OF NII RESOURCES (FROM 1987)

<table>
<thead>
<tr>
<th>FINANCIAL YEAR</th>
<th>EFFORT(^1)</th>
<th>NO. SITE VISITS(^2)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1987/1988</td>
<td>11.7</td>
<td>5</td>
</tr>
<tr>
<td>1988/1989</td>
<td>8.2</td>
<td>5</td>
</tr>
<tr>
<td>1989/1990</td>
<td>11.9</td>
<td>11</td>
</tr>
<tr>
<td>1990/1991</td>
<td>12.3</td>
<td>20</td>
</tr>
<tr>
<td>1991/1992</td>
<td>15</td>
<td>22</td>
</tr>
<tr>
<td>1992/1993</td>
<td>16.7</td>
<td>33</td>
</tr>
<tr>
<td>1993/1994</td>
<td>17</td>
<td>38</td>
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<tr>
<td>1994/1995</td>
<td>11.2</td>
<td>44</td>
</tr>
<tr>
<td>1995/1996(^3)</td>
<td>3.2</td>
<td>20</td>
</tr>
<tr>
<td>Total</td>
<td>107.2</td>
<td>198</td>
</tr>
</tbody>
</table>

Notes:

1. Effort is given in staff years.

2. This is the number of separate visits to site each year, the number of inspectors involved and the duration of the visit varied depending upon the nature of the inspection.

3. Figures for 1995/1996 are for the first two quarters of the financial year (April to September 1995)
<table>
<thead>
<tr>
<th>LASP</th>
<th>TOPIC</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Radiological Protection</td>
</tr>
<tr>
<td>2</td>
<td>Human Factors</td>
</tr>
<tr>
<td>3</td>
<td>Structural Integrity</td>
</tr>
<tr>
<td>4</td>
<td>Station Operating Instructions and Technical Specifications</td>
</tr>
<tr>
<td>5</td>
<td>Internal Hazards and External Hazards</td>
</tr>
<tr>
<td>6</td>
<td>Fault Analysis (PSA) and Human Factors in Fault Analysis (HFFA)</td>
</tr>
<tr>
<td>7</td>
<td>Transient Analysis</td>
</tr>
<tr>
<td>8</td>
<td>Simulator Training and Operator Training</td>
</tr>
<tr>
<td>9</td>
<td>Radioactive Waste</td>
</tr>
<tr>
<td>10</td>
<td>Primary Protection System (PPS), Secondary Protection System (SPS) and WISCO</td>
</tr>
<tr>
<td>11</td>
<td>Electrical Systems and Mechanical Systems</td>
</tr>
<tr>
<td>12</td>
<td>Quality Assurance</td>
</tr>
<tr>
<td>13</td>
<td>Civil Structures</td>
</tr>
<tr>
<td>14</td>
<td>Commissioning</td>
</tr>
<tr>
<td>15</td>
<td>Pre-Operational Safety Report (POSR)</td>
</tr>
<tr>
<td>TYPE</td>
<td>MAIN BUSINESS</td>
</tr>
<tr>
<td>-----------</td>
<td>-------------------------------------------------------------------------------</td>
</tr>
<tr>
<td>Level 1</td>
<td>• Strategic plans and priorities</td>
</tr>
<tr>
<td></td>
<td>• Safety and regulatory policy and strategy</td>
</tr>
<tr>
<td></td>
<td>• Consider matters raised from Level 2</td>
</tr>
<tr>
<td></td>
<td>• Monitor overall progress</td>
</tr>
<tr>
<td>Level 2</td>
<td>• Implement safety and regulatory strategy</td>
</tr>
<tr>
<td></td>
<td>• Plan for key stages in licensing programme</td>
</tr>
<tr>
<td></td>
<td>• Consider major regulatory/safety issues</td>
</tr>
<tr>
<td></td>
<td>• Consider matters raised from Level 3</td>
</tr>
<tr>
<td>Level 3</td>
<td>• Discuss and agree programmes of work</td>
</tr>
<tr>
<td></td>
<td>• Progress resolution of regulatory issues</td>
</tr>
<tr>
<td></td>
<td>• Progress resolution of specific issues</td>
</tr>
<tr>
<td></td>
<td>• Consider matters raised from Level 4</td>
</tr>
<tr>
<td></td>
<td>• Formalise agreements</td>
</tr>
<tr>
<td>Level 4</td>
<td>• Gathering and exchange of information</td>
</tr>
<tr>
<td></td>
<td>• Testing of understanding</td>
</tr>
<tr>
<td></td>
<td>• Identify issues to be raised at Level 3</td>
</tr>
<tr>
<td>NO.</td>
<td>TITLE</td>
</tr>
<tr>
<td>-----</td>
<td>-------------------------------</td>
</tr>
<tr>
<td>1</td>
<td>Interpretation</td>
</tr>
<tr>
<td>2</td>
<td>Marking of the Site Boundary</td>
</tr>
<tr>
<td>3</td>
<td>Restriction on Dealing with the Site</td>
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<tr>
<td>4</td>
<td>Restrictions on Nuclear Matter on the Site</td>
</tr>
<tr>
<td>5</td>
<td>Consignment of Nuclear Matter</td>
</tr>
<tr>
<td>6</td>
<td>Documents, Records, Authorities, Certificates</td>
</tr>
<tr>
<td>7</td>
<td>Incidents on Site</td>
</tr>
<tr>
<td>8</td>
<td>Warning Notices</td>
</tr>
<tr>
<td>9</td>
<td>Instructions to Persons on Site</td>
</tr>
<tr>
<td>10</td>
<td>Training</td>
</tr>
<tr>
<td>11</td>
<td>Emergency Arrangements</td>
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<tr>
<td>NO.</td>
<td>TITLE</td>
</tr>
<tr>
<td>-----</td>
<td>-------</td>
</tr>
<tr>
<td>12</td>
<td>Duly Authorised and Other Suitably Qualified and Experienced Persons</td>
</tr>
<tr>
<td>13</td>
<td>Nuclear Safety Committee</td>
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<tr>
<td>14</td>
<td>Safety Documentation</td>
</tr>
<tr>
<td>15</td>
<td>Periodic Review</td>
</tr>
<tr>
<td>16</td>
<td>Site Plans, Designs and Specifications</td>
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<td>17</td>
<td>Quality Assurance</td>
</tr>
<tr>
<td>18</td>
<td>Radiological Protection</td>
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<tr>
<td>19</td>
<td>Construction or Installation of New Plant</td>
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<td>20</td>
<td>Modification to Design of Plant Under Construction</td>
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<td>21</td>
<td>Commissioning</td>
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<td>22</td>
<td>Modification or Experiment on Existing Plant</td>
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<tr>
<td>NO.</td>
<td>TITLE’</td>
</tr>
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<td>-----</td>
<td>--------</td>
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<tr>
<td>23</td>
<td>Operating Rules</td>
</tr>
<tr>
<td>24</td>
<td>Operating Instructions</td>
</tr>
<tr>
<td>25</td>
<td>Operating Records</td>
</tr>
<tr>
<td>26</td>
<td>Control and Supervision of Operations</td>
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<tr>
<td>27</td>
<td>Safety Mechanisms, Devices and Circuits</td>
</tr>
<tr>
<td>28</td>
<td>Examination, Inspection, Maintenance and Testing</td>
</tr>
<tr>
<td>29</td>
<td>Duty to Carry Out Tests, Inspections and Examinations</td>
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<td>30</td>
<td>Periodic Shutdown</td>
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<tr>
<td>31</td>
<td>Shutdown of Specified Operations</td>
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<td>32</td>
<td>Accumulation of Radioactive Waste</td>
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<td>33</td>
<td>Disposal of Radioactive Waste</td>
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<td>34</td>
<td>Leakage and Escape of Radioactive Waste</td>
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### Table 7 continued - standard licence conditions

<table>
<thead>
<tr>
<th>NO.</th>
<th>TITLE(^1)</th>
<th>PURPOSE(^1)</th>
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</thead>
<tbody>
<tr>
<td>35</td>
<td>Decommissioning</td>
<td>To require licensee to make adequate provisions for decommissioning to give discretionary powers to NII(^2) to direct decommissioning be commenced or halted</td>
</tr>
</tbody>
</table>

**Notes to Table 7:**

1. Under each licence condition there are one or more clauses which stipulate the particular requirements. Details of the requirements and further explanation of their purpose are given in the HSE publication "Nuclear Site Licences under the Nuclear Installations Act 1962 (as amended) - Notes for Applicants" ISBN 0-7176-0795-X.

2. The licence conditions refer to the Health and Safety Executive. NII is responsible for enforcement of the licence and its conditions on behalf of the Executive.

3. Nuclear matter other than excepted matter or radioactive waste.

4. Relevant site as defined by the Nuclear Installations Act 1965.
FIGURES
Figure 2 Nuclear fuel assembly
Figure 3 Reactor coolant system
Figure 4 Reactor pressure vessel and internals
Figure 6 Steam generator
Figure 9 Station control and protection systems

Legend
- These systems are collectively known as the data processing and control systems (DP/C).
- Dashed support rather than control. The technical support centre contains no controls.
Figure 10 Reactor trip system-voting logic arrangement
FIGURE 11
REMOVED
Figure 12 Main steam system (one leg)
Figure 13

REMOVED