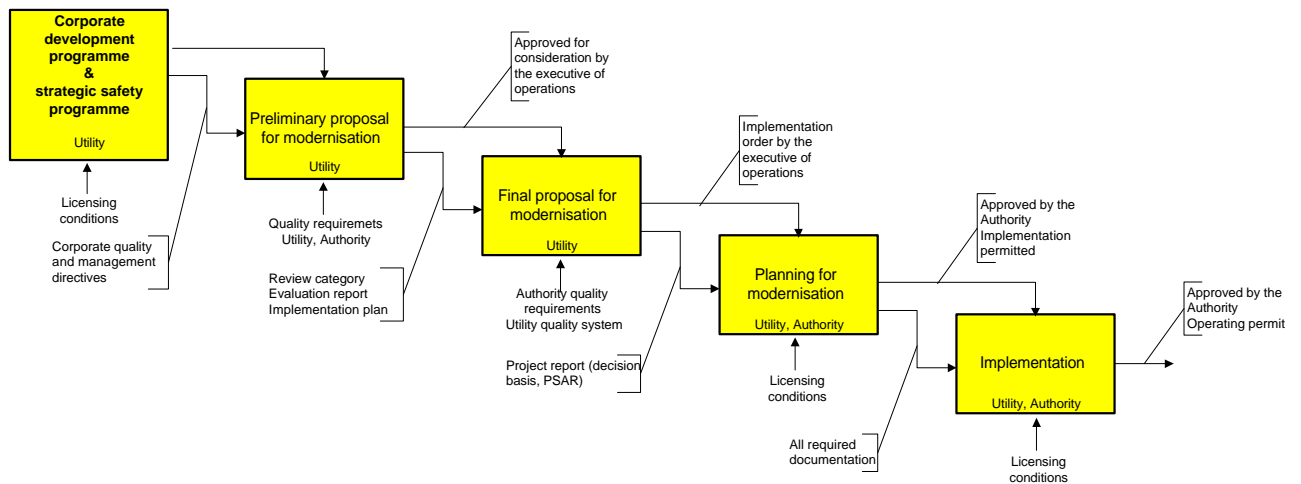


Modernisation for Maintaining and Improving Safety at Nordic Nuclear Power Plants

Lennart Hammar¹, Björn Wahlström and Kaisa Simola²

February 1998



¹ ES-Konsult AB, Stockholm

² Technical Research Centre of Finland (VTT)

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Rapporten er udgivet af:
NKS-sekretariatet
Bygning 100
Postboks 49
DK-4000 Roskilde

Tlf. +45 4677 4045
Fax +45 4677 4046
E-mail annette.lemmens@risoe.dk
<http://www.risoe.dk/nks>

Modernisation for Maintaining and Improving Safety at Nordic Nuclear Power Plants

A report from the Nordic Nuclear Safety Programme (NKS)

Project NKS/RAK-1.5

by

Lennart Hammar

ES-Konsult AB, Stockholm

Björn Wahlström and Kaisa Simola

Technical Research Centre of Finland (VTT)

February 1998

Abstract

The safety practices in Finland and Sweden are described and compared in regard of effecting modernisation for safety of the nuclear plants in the two countries, considering new technology and advancing safety requirements as proposed for new reactors. Particular attention is given to strategies for applying new safety requirements to reactors built to earlier standards, and to the interplay between the nuclear utilities and the safety authorities.

Overviews are given of past and current modernisations of the nuclear power plants in Finland and Sweden.

The management procedures in controlling the implementation of modifications to the nuclear power plants are described and discussed in regard of prevailing differences between Finnish and Swedish practices. A formal modelling technique (SADT) was applied for capture of the essential contents of the relevant documented procedures.

Two examples of recent plant modifications in the Finnish nuclear plants in Olkiluoto and Loviisa are described and discussed in greater detail.

Recommendations are given.

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1 INTRODUCTION

The study reported below (RAK-1.5) formed part of a research project, "Strategy for Reactor Safety" (RAK-1), conducted under the Nordic Nuclear Safety Programme (NKS) which is sponsored for main part by the Nordic radiation protection and nuclear safety authorities and partly by the nuclear industry in Finland and Sweden. This project, RAK-1, consisted of five studies in total, of which one (RAK-1.1) aimed at mapping and assessing the safety work in general and the four others at examining issues of particular interest in some further depth [ref. 1]. Particular weight was assigned to identifying and analysing notable differences between the practices in Finland and Sweden, and between different power plants, in an attempt to find out what lessons could possibly be learned in this way.

The RAK-1.5 study was aimed at assessing and comparing strategies and practices at the nuclear utilities and the authorities in Finland and in Sweden for ensuring that the nuclear plants keep reasonably well in pace with the development in regard of safety, considering new knowledge and technology, operating experience and generally advancing safety requirements.

The important matter of ensuring that the full potential of the plant modifications in regard of safety can also be gained in practice, as intended, by proper quality assurance, was also covered in the study.

As indicated by its title, the present report is mainly concerned with *modernisation for safety*, i.e. making plant modifications to achieve safety improvements, beyond the safety requirements in the current operating licenses, by taking additional safety measures in regard of systematic feed-back of experience, modern technology and modern safety practices. Modernisation mainly for improved output and productivity and for correcting design faults, which have been revealed by incidents or accidents rather than in proactive safety research and analysis, as well as mere renovation of degraded plant equipment to match the original requirements in regard of performance and safety thus are not specifically discussed in this report.

The approach made in this study was to view the past and the current modernisation activities in the operating nuclear power plants in Finland and in Sweden in relation to the safety standards set by the licensing conditions, the considerable operating experience gained through the years, the progress made in science and technology, and the enhanced safety goals currently applied. Information and views were obtained by means of questionnaires and interviews at the plants and at the safety authorities.

For expressions followed by *, please refer to Appendix 1: Explanation of some terms.

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2 THE DEVELOPMENT OF SAFETY

2.1 Driving forces in regard of modernisation for safety

In most fields in society and industry in the developed countries, there is a growing concern about matters of risk and safety and about the environment. In the public opinion nuclear power is, for the time being, a safety concern. Reducing this, as may be possible, e.g., by modernisation, is clearly for large part driving the development.

The *worth of the vast investments* in the nuclear plants is in itself another important driving force, going far if not completely along with the concern for safety. This provides a particular incentive on part of the plant owners, in addition to their natural concern about the safety, to avoid losing their plants or the permits to operate them due to accidents. It also provides particular incentive to counteract degradation by ageing and obsolescence, so as to extend the useful life of the plants, again as well in the interest of safety. Incidentally, obsolescence is a concern not only in regard of operation and maintenance of the plants but potentially also in regard of safety. It may thus eventually lead to unavailability of vital spare parts and rendering operation and maintenance more difficult and costly, in part due to decreasing familiarity of the plant staff with the obsolete equipments or systems.

The business related incentives exist of course only as long as the operation of the plants remains competitive and the plants are not approaching to require major overhaul and upgrading - an important aspect to be observed in advance by the authorities. A particular problem in Sweden is that the business related incentives obviously vanish as the operation of the plants approaches the time of final closure.

The tendency towards *shrinking safety margins due to the strive for enhanced output and productivity* of the nuclear plants, is a matter of concern contributing further to driving the safety requirements. Power up-ratings of the reactors are common examples.

Improved knowledge and insight, as can be gained by *experience, analysis, research and development*, are major driving factors. Systematically gathered operating experience thus provides, upon systematic safety analysis, increasing insights in regard of possible risks and possibilities as for safety improvements. All minor incidents and occurrences are important in this regard, as any weaknesses they may reveal would be likely to contribute as well to the risks for more serious occurrences and accidents.

The extensive research in the field of nuclear safety is important in providing knowledge about possibly unknown risks and phenomena relevant to the risks and paves the way for inventing and developing novel safety features.

The improved knowledge and insight gained through the years not only concern technical matters but also "soft" factors, such as human behaviour and the interplay between man, technology and organisation (MTO, Man-Technology-Organisation).

The implementation of new technology for improving the safety raises on the other hand, in itself, questions about possibly associated, new risks, and how to ensure that the expected safety benefits will be gained in actual practice. The advanced software based digital control systems, offering not only great potentials but in addition some unusually intricate quality assessment problems, is a good example.

The design, construction, maintenance and operation of the nuclear plants thus all are subject to fairly frequent modifications. Obviously, for any attempted modification, careful consideration of the safety implications will be necessary to ensure, at the very least, that safety will not be challenged. On the other hand, attempted modifications of the nuclear plants, e.g. for enhanced performance or economy, offer additional opportunity for pursuing improvement of the safety, partly as the safety authorities will then become suitably involved in their capacity of granting required permits.

2.2 Safety criteria

2.2.1 Safety goals and approaches

Safety criteria consist partly of prescribed safety goals and partly of prescribed safety approaches (pertaining to design, construction, operation, maintenance etc.) for achieving the safety goals in practice. There have been important developments in both regards during the years.

The *safety goals* have become more clearly defined by using "probabilistic targets"¹ for the risk reduction, in regard of core damage accidents as well as of accidental releases of radioactivity to the environment in excess of tolerable limits. The *scope* of the safety goals has, furthermore, been extended to cover more completely the possible risks, i.e. by requiring that all possible initiating events and hazards are indeed properly accounted for in the safety assessment. Examples will be given in the following (section 4.2.1, page 18).

The requirements in terms of prescribed *safety approaches* and methods for achieving the safety goals have, in addition, become increasingly stringent. This applies to all aspects of the plant safety including design, construction, operation and maintenance.

The main rationale for relying on rules to ensure safety is that the rules embody the established experience and knowledge, the state of the art, and that they facilitate the practice of it. Too much reliance on rules would, however, hinder the development.

The demands for credible *verification of the safety* have as well increased markedly. This applies to founding the assumptions and the modelling applied in the safety analysis on truly established science and technology. A concept of *robustness* of the safety case has been coined, by which is meant that there should be no nearby "cliff edge" due to the safety case being overly sensitive to some vital factor or assumption. Any possible uncertainties should be accounted for in a conservative manner and well established, stringent procedures should be used to ensure firm and complete bases for the final safety assessment. Finally it is required that it be checked from time to time that the verification of the safety continues to hold in view of new knowledge.

Uncertainty in the assessment of the safety of any given plant or system obviously implies *reduced rating of its safety*. The safety may thus be enhanced by advancing the safety analysis, i.e. making use of more advanced methods and new knowledge, an approach normally preferable to just adding new safety features. The uncertainties may also be reduced by means of improved *quality assurance* and *qualification* of the safety features, i.e. to ensure that they are indeed qualified in all regards to serve their intended purposes, e.g. in regard of withstanding environmental conditions which may prevail

¹ Terms with asterisk (*) refer to Appendix 1: Explanation of some terms

under accident conditions. Uncertainty with respect to actual plant conditions can be reduced with *inspections* and by performing *tests* on the components.

2.2.2 Rulemaking and international conventions

International co-operation and consensus has played great part in the development of currently applied requirements for safety.

The rulemaking and the safety practices were in most countries, from the outset, largely based on the American rulemaking, having developed - since the late sixties - into the present General Design Criteria, i.e. Appendix A of the U.S. Code of Federal Regulations (10 CFR 50), together with its other relevant Appendices and some 170 associated Regulatory Guides. These regulations still provide important guidance for national rule-making.

Internationally agreed guidance for the national reactor safety rulemaking, pertaining to siting, design, operation and quality assurance, is available since the mid seventies in the international IAEA Nuclear Safety Standards (NUSS) for Nuclear Installations with supplementing Safety Guides and Safety Practices. The top tier *safety principles*, contained in the NUSS top document, *Safety Fundamental - The Safety of Nuclear Installations* [ref. 2], have been largely adopted in the Nuclear Safety Convention [ref. 3] which became legally binding for its parties, including all Nordic countries, when entering into force in October 1996.

The current nuclear safety regulation and corresponding international guidance is rather slow to entirely reflect the increasing demands for safety, particularly as it is bound to allowing reasonable terms for the growing number of operating plants built to earlier standards. A better reflection is seen in the extensive discussion on nuclear safety matters and safety research, e.g. in the IAEA, the OECD/NEA and the EC. This has provided important input in establishing stricter requirements for the upgrading of the safety of the operating reactors as well as new requirements to be applied for future reactors.

Impetus to the development of the safety requirements has, for example, been effected by reports by the distinguished IAEA International Senior Advisory Group (INSAG). The document *Basic Safety Principles* (INSAG-3, 1988) [ref. 4] thus forms the basis of the aforementioned NUSS Safety Fundamentals. Another document, *A Common Basis for Judging the Safety of Nuclear Power Plants Built to Earlier Standards* (INSAG-8, 1995) [ref. 5], provides practical guidance in requiring modernisation of nuclear plants on the basis of modern safety criteria.

2.2.3 Utility requirement standards

In the USA, the utility owned Electric Power Engineering Institute (EPRI), developed during the years 1985-1995 the Utility Requirements Document (URD) [ref. 6]. The European utilities started in 1992 to work out a similar document, the European Utility Requirements (EUR) [ref. 7]. The EUR are addressed to the designers and suppliers of the next generation of light water cooled (LWR) plants with the primary objective of *harmonising*

- the safety targets and the methods for assessing the safety;
- the requirements for performance, reliability, durability and profitability of the nuclear plants;
- the corresponding requirements on design and construction, considering also the final decommissioning.

Expected benefits are:

- improved conditions for licensing, public acceptance, and for countries with nuclear power to appear as "good neighbours", as can be achieved by unified and consistent approach;
- improved competitiveness of nuclear energy as can be achieved by standardisation, simplification, enhanced availability, extended lifetime of the plants as well as by stable conditions for the competition between the suppliers.

The safety objective of the EUR is to ensure compliance with the licensing requirements in all member countries without significant needs for adjustments.

Probabilistic targets are set for core damage (10^{-5} per reactor year) well as for exceeding fixed limiting off-site releases (10^{-6} per reactor year). The objective in regard of off-site releases is that there should be no off-site consequences from any severe accidents, e.g. in terms of evacuation or sheltering. Limiting releases for accident conditions (design basis) and "design extension conditions" are defined accordingly.

The requirements put great emphasis on "forgiving behaviour" of the plant, ample safety margins, generous grace times for any required operator actions, defence-in-depth* based on independence of the activity barriers and the defence lines. Highly reliable containment function in severe accident situations is to be achieved by design provisions aimed at excluding high pressure core melt scenarios and accident sequences involving bypassing, hydrogen explosions or steam explosions. The integrity of the containment must be ensured even under extreme severe accident loads (e.g. global deflagration of the maximum amount of hydrogen) without active containment heat removal during at least 12 hours. Double containment is, in fact, prescribed.

Since 1996 the Finnish and the Swedish nuclear utilities take part in drafting the EUR documents together with, at present, the power or nuclear power utilities in Belgium, France, Germany, Italy, Netherlands, Spain and UK.

The present state of the development of the EUR is that the Volume 1, containing the main policies and the top tier requirements, and the Volume 2, containing the generic requirements, are both currently on a second round for comments. Volume 3, addressing the specific requirements for particular plant designs, including the NPI European Pressurised Reactor (EPR) and the ABB Atom reactor BWR-90, is under preparation. There is also a Volume 4 under preparation, dealing with the conventional power generation parts of nuclear plants. It may be noted that the EPRI URD and the IAEA NUSS documents have been duly considered in developing the EUR in order to ensure consistency.

2.3 Advanced safety concepts

Advanced technology for safety, as may be applied in new reactors, is of interest as being for some part applicable also for modernisation of the older reactors. The level of safety

claimed for advanced reactors will in any case largely reflect on the safety requirements being applied to the older reactors.

A number of advanced LWR (Light Water Reactor) concepts, presently under development, reflect the advancing technology for LWR nuclear safety. There are so-called evolutionary concepts on one hand and the more radically innovative on the other, representing two somewhat different approaches. The evolutionary approach aims at consolidating the design on a step-by-step basis, so as to allow building extensively on operational experience in each step.

The other approach, sometimes called "revolutionary", obviously relies more on experiments and prototypes for demonstration of the safety and the performance of the concept. Whilst both approaches employ "passive" safety* features, the innovative concepts usually go far in this respect. Although the potential for safety thus offered may indeed be high, careful verification is nevertheless required that they will at least be as safe as the traditional active safety systems* [ref. 8].

The evolutionary concepts appear at present as the most viable in regard of possibly becoming realised commercially in a foreseeable future. Their general safety features would be assumed to be largely in accordance with the EUR (cf. above, section 2.2.3, page 7).

Evolutionary, next generation LWR concepts include the European Pressurised Reactor (EPR), serving as a test concept for the EUR [ref. 9] and the ABB Atom BWR 90+, also to be covered in Volume 4 of the EUR (cf. section 2.2.3, page 7). For safety approaches taken in a number of advanced LWR concepts, including the implementation of passive safety features in the design, cf. [ref. 10,11]. A comprehensive catalogue of concepts and designs of advanced reactors are given in [ref. 12].

So-called Final Safety Evaluation Reports (FSER) have been completed by the U.S. NRC in 1994 for the ABB Combustion Engineering System 80+, an advanced PWR [ref. 13], and for the General Electric ABWR, an advanced BWR [ref. 14]. A similar report for the Westinghouse Electric AP 600, an advanced PWR, is expected to be completed during 1997.

2.4 The scope of safety modernisation of ageing plants

The operating reactors in the Western world are typically subject to modifications and backfitting from time to time, reflecting lessons learned through experience feed-back and safety analysis, e.g. as can be seen in a recent compilation made for the SKI [ref. 15].

Modernisation for safety aims at enhancing the defence-in-depth* system, including:

- *the first line of defence* (the preventive defence line), i.e. the inherent safety and the quality of the plant activity barriers* and the plant processing and control systems, aimed at minimising the challenges presented to the supplementary lines of defence;
- *the supplementary lines of defence* (the reactive defence lines), i.e. the reliability and the efficiency of the safety* and mitigatory* systems in performing their intended functions;
- *the completeness* of the defence-in-depth* system, for ensuring the availability of all reasonably required provisions;

Typical examples of modernisation, *relating to the first line of defence*, include measures and modifications to reduce the probability of pipe breaks causing LOCA*, particularly accidents involving large losses of primary coolant through leaks at locations below the reactor involving major challenges to the safety systems*.

Modernisations of the *safety and mitigatory systems*, providing for the supplementary lines of defence, typically belong to some of the following categories:

- Enhanced redundancy* and diversity*, e.g.:
 - added redundant and diverse safety trains*, e.g. for emergency core cooling and containment spray;
 - added diversity* to highly redundant* safety systems* such as the scram system and the RPV pressure relief and automatic de-pressurisation system;
 - added redundancy and diversity to reactor protection systems and safety related monitoring (e.g. RPV level);
 - added safety system redundancy and diversity by system cross-connections, enabling safety systems as well as their water and power supplies etc. to serve diverse purposes;
 - use of mobile equipment for emergency supply of power and water.
- Improved protection of the systems against internal and external hazards, e.g.:
 - against fire, flooding and consequential damages from initiating events, such as missiles generated in pipe ruptures, by stricter separation* of the safety trains*, including their controls and supplies;
 - against fire by means of, in addition, enhanced fire alarm and fire fighting systems
 - against external events by bunkering of vital safety systems* and supplies;
- Improved performance and autonomy* by, e.g.:
 - extended automation for improved working conditions of the plant personell;
 - enhanced operator aids, notably in the control rooms;
 - enhanced control system logic (actuation of the safety functions, automatic protection, interlocking etc).;
 - adoption of new technology for enhanced performance (e.g. in supervision and control systems);
 - application of advanced safety concepts (e.g. ”passively*” actuated safety systems*).

The development, in the aftermath of the accident in TMI-2, of provisions to mitigate severe core damage accidents can be seen as a contribution enhancing the *completeness of the defence-in-depth* system*. Other contributions concern coping with external events as seismicity, explosions and airplane crashes.

3 GENERAL STRATEGIES FOR SAFETY MODERNISATION

In order to form a view of the main strategies applied by the utilities and the authorities in Finland and Sweden in regard of modernisation, enquiries and interviews were conducted considering mainly:

- goals and policies, and how they are communicated;
- general incentives for modernisation, "philosophy";
- specific causes for initiating past and currently planned modernisations;
- the interplay between the utilities and the safety authorities.

3.1 The corporate view

Modernisation, a necessary pre-requisite for staying in business - also in regard of the developing safety requirements - is a concern firstly and mainly on part of the utilities. Obviously, great need is felt of being in reasonable control of the future of the business. For large part this is a matter of the nuclear utilities maintaining their plants at highest possible performance/cost ratio by farsightedly counteracting on deterioration by ageing and obsolescence. Not least, it is a matter of coping with increasing requirements for safety as well as for performance.

The corporate strategy will have to account for visions extending something like 10-20 years into the future. The visions will then, from time to time, be turned into strategic plans in a medium time perspective, e.g. up to five years.

The end foreseen for using nuclear power in Sweden constitutes in itself a major uncertainty. The corresponding strategy is clear on part of all Swedish utilities, however: *managing the nuclear business for the considerable time it will still be filling an urgent need in Sweden - now foreseen to be beyond the year 2010 - is incompatible with any strategy other than essentially disregarding the intended phasing out of it.*

Safety, as well as gaining confidence among the public for minding about the safety, is a main utility concern. The various aspects have been largely dealt with in other sections of this report, e.g. in regard of the driving forces (section 2.1, page 5). The safety concerns on part of the utilities should be largely reflected in the objectives set for the EUR (section 2.2.3, page 7).

Major considerations thus include the ageing of the plants. Degradation of the nuclear plant components due to wear and ageing usually cause less need for renewal than obsolescence. Many components have an *economically useful life* which is shorter than their *possible service life*, if obsolescence would not cause ever increasing costs and shortage of spare parts, growing incompatibility with other technology and inefficiency as compared with using modern alternatives, as well as deteriorating familiarity of new generations of staff with old technology.

On the other hand, in regard of the potential economic lifetime of the nuclear plants, there are few systems and components, even major ones, which could not be replaced for efficient renewal of the plants, provided of course that the business is viable enough to sustain the necessary investments. The renovation and modernisation of the Oskar-

shamn 1 reactor constitute convincing demonstration, having attracted world-wide attention.

In the utility perspective it is particularly important to remain in control of the situation in regard of upcoming safety issues and new demands, to avoid instances of the authorities taking complete charge. One important means is to ensure, from the outset, adequate *safety robustness* - i.e. ample margins in complying with currently applied safety requirements (cf. section 2.2.1, page 6) - as essential in order to cope well with changing and enhanced requirements.

3.2 Main corporate strategies

If not always made fully clear in the past, all utilities rank presently safety highest in stating their general performance and business goals. Typically, all these goals, documented in well established top tier documents as "Management and Quality handbooks", are actively communicated in the organisations to guide the operation and the development of the nuclear plants. Each utility specifically state its safety goals, mainly in accordance with chapter 4.2, together with the operational goal of always aiming for improvement of the safety (4.1.1).

Modernisations are typically implemented in conjunction with long range investment programmes and managed in firmly established frameworks of well coordinated tasks encompassing major functional units of a plant. Such approach facilitates coordination of all detailed requirements in regard of performance as well as verification of the over-all consistency and functionality of the total concept by means of common testing programmes. In this way it is also easier to provide for consistent and facilitated operating procedures and proper training of the staff as required.

The strategies for modernisation in regard of safety consist for large part in establishing the required bases, as described in section 4.1. While, in Sweden, the special design basis reviews play currently a particularly important role, corresponding purposes are considered in Finland to be achieved in more regular ways (section 4.1.4).

As modernisation usually involves introducing new technology and approaches, communication with the safety authorities on a long term in advance is obviously important. It is on the other hand important that clear and predictable development of the regulatory requirements and criteria can be foreseen. An open attitude to concerns and desires on part of the authorities is promoted, while emphasising the role of the utilities in proposing, as responsible for the safety, systematic and consistent safety concepts for final approval of the authorities.

3.3 The part played by the safety authorities

General aspects in regard of the interplay between the safety authorities and the utilities have been covered in the RAK-1.1 project, "Strategy for Reactor Safety" [ref 1].

The safety goals on part of the authority, in regard of modernisations, are no others than those generally applied by the safety authorities (cf. section 4.1.1, page 14). The main strategy of the authorities is to *proactively* promote continuous improvement of the safety by repeatedly questioning the current safety level in relation to current safety standards and technology. Particularly good opportunities are offered in conjunction with

modification or renovation projects, as associated safety matters are then usually brought to light.

This strategy, to be pursued by the safety authority, obviously requires that the authority will be active in gaining, by suitable inspections and safety assessment programmes, as good knowledge of the safety of the plants as possible in regard, particularly, of weaknesses and corresponding potentials for improvements. It also requires the general competence of the authority to be developed in all related areas, e.g. by research and international cooperation.

Depending on the regulatory policy, the safety authority may be involved in the review and licensing of modifications to a varying degree. An important consideration, in addition to ensuring safety, is that the authorities need to take active part in providing for a well founded and consistent approach to improving and modernising the nuclear plants in a long range perspective. Another consideration relates, again, to the need on part of the authorities to develop their competence.

The safety authorities strive, however, at playing their part in the modification processes so as not to detract from the incentive on part of the utilities in assuming their full responsibility for the safety. They will also have to consider their own resources, in which respect the conditions in regard of the Swedish authority are the less favourable. In relative terms, the efforts spent on part of the authority, in terms of reviews and inspections on the sites, are thus greater in Finland than in Sweden, covering in Finland also smaller modifications of lesser safety significance, and generally going more in technical detail.

There is, on the other hand, an outspoken strategy of the Swedish safety authority to gain the required assurance of the safety for appreciable part by assessing the quality of the modification process, as being conducted by the utilities, and to lesser extent by taking direct part in double-checking on technical matters. Nevertheless, the Swedish authority will usually be heavily involved in significant modification projects including the technical matters.

In considering the various safety issues, the safety authorities in both countries strive at reaching, to the extent possible, mutual understanding and consensus with the utilities as to what is in fact required to achieve an acceptable level of safety. The rationale is to ensure maximum benefit from all available expertise. In addition to the formal means of communicating between the safety authorities and the utilities, there is accordingly great emphasis on maintaining sufficient informal contacts at various management levels, e.g. regular annual or biannual top management meetings, and meetings concerning on-going modification projects. As another example, there is an informal contact group of experts from the SKI and all Swedish utilities which meets regularly to discuss major safety issues on a long range perspective

”The Swedish model” is an actively pursued notion in Sweden. According to this model, the safety authority is expected to promote its safety goals so as to first of all allow drawing on the willingness, the competence and the ability offered by the utilities - being responsible for the safety - for reaching the goals. The aims of the safety authority are similar in Finland, but the working model differs in that the authority requires, e.g., proof and evidence of proposed plant modifications to be presented for approval in practically all cases rather than in selected cases, as practised in Sweden.

It is anyhow well realised, on part of both safety authorities, that the safety authority should avoid stressing its particular views or concerns in a way to hinder the utilities to systematically develop consistent safety concepts.

4 SAFETY MODERNISATION IN FINLAND AND SWEDEN

4.1 Practices in establishing bases for modernisation

4.1.1 Continuous upgrading of the licensing basis

The licensing conditions of the operating reactors, as presented by the utilities and approved by the authorities, are documented in the final safety reports (FSAR). An FSAR is regarded as representing the *minimum* safety level the licensee has contracted to maintain as a condition for the permission to operate the reactor.

In Finland, as well as in Sweden, the FSAR is a "living document", subject to updating to always reflect current status in regard of any modifications of the plant or the way it is operated. The licensees are required by the authorities to consider, at all times, making safety improvements *whenever appropriate and reasonably achievable*. The idea of considering safety improvements at all times might appear unreasonable. The rationale, however, is to make efficient use of all opportunities offered for safety improvements, as, e.g., in conjunction with repairs or re-constructions. Besides, taking at any time the view that safety is adequate would indeed imply complacency and decline of the safety culture.

Finland

According to Finnish general regulations for the safety of nuclear power plants [ref.16], the adequacy of the plant safety and the licensing basis shall be continuously re-evaluated on the basis of operating experience as well as research and development, including methods used for assessment of safety. Safety criteria have been established by the Finnish safety authority (STUK) in the so-called YVL safety guides [ref. 17, 18]. The YVL-guides reflect the currently intended level of safety, and they are continuously subject to update. If the updating includes new safety requirements, their application on operating plants is specifically decided by the STUK in each case.

The updating of the YVL-guides follows a strict procedure calling for, among other things, thorough expert review by an Advisory Committee on Nuclear Safety designated by the Finnish safety authority.

In order to keep, at all times, a valid view of the safety implications of the advances made in research and technology, and of changing practices in applying safety requirements, the STUK maintains an ongoing inspection and review programme in which selected issues of particular importance are systematically covered on a yearly basis [ref. 18]. The programme also aims at assessing the quality and the development of the safety related activities at the utilities. If necessary, the assessments are made known to the licensee for possible actions. The issues brought up may concern possibilities of safety improvements on the basis of advancing technology, new information and research, or changed safety requirements.

Sweden

In Sweden, the nuclear safety legislation just requires that safety be provided for, leaving it to the safety authority, the SKI, to judge what is safe enough and to apply, for each operating licence, any detailed conditions accordingly required. General safety goals have thus been defined by the SKI, in harmony with the Nuclear Safety Convention to which Finland as well as Sweden are committed, which are documented in SKI's yearly Activity Plan, subject to Government approval. The General Design Criteria in the U.S. Code of Federal Regulations (10 CFR 50), and the Regulatory guides issued by the U.S. NRC, together with the NRC Standard Review Plan [ref. 19], have contributed valuable guidance in establishing the safety requirements embodied in the FSARs of the Swedish reactors.

Nevertheless, there have been needs felt to improve the consistency of the safety requirements as can be effected by establishing domestic safety rules to apply in general. Such general rulemaking is in progress since 1993 when the SKI was granted the required legal competence by a modification of the nuclear safety legislation. The regulations issued by SKI (e.g. those issued so far which concern mechanical and pressure vessel components, ref 32) differ in from the Finnish YVL-guides in constituting strict legal requirements, although exemptions may in special cases be granted by the SKI provided the aims of the Law of Nuclear Activities are not set aside.

Like the Finnish safety authority, the SKI keeps important safety issues at the plants under close supervision, as well as the general development of related matters, by means of dedicated safety reviews and inspections. Moreover, attention is paid to the current development of safety standards for new reactors, as they are likely to apply to all operating reactors in the long run. Thus, a special project, the so-called R-2000 [ref 20] is presently under way at the SKI for establishing proper bases for its future work in promoting the safety goals, particularly in conjunction with a growing number of plant modifications, as being expected in the next coming years. An important aim of the Project R-2000 is to enable clear indications to be given to the plant owners about the position in this regard to be expected on part of the SKI.

In a first stage the R-2000 project aims at identifying safety requirements, in regard of initiating events and related safety functions, which are likely to become enhanced in regard of current safety practices, and would require the plants to be modified at significant costs.

4.1.2 Relicensing

In Finland, the operating licenses are typically granted for 10 years. The required renewal of the licenses in Finland provides for thorough re-assessment of the overall safety of the nuclear plants, largely on the basis of a compilation of the inspection and assessment work done during the previous license period. The license renewals thus serve as an important complement to the ongoing inspection and assessment activities.

Since the legislation requires continuous updating of all important documents concerning the safety, e.g. the FSAR, the PSA, and the Technical Specifications for operation, only recently changed documents are required to be submitted with the application for a renewal of the license.

The present operating licenses of the Finnish nuclear power plants are due for renewal by the end of 1998. STUK has prepared an Action Plan, in which requirements to be met for obtaining renewal of the operation have been compiled. Reports and analyses are thus required for assessment of the overall safety and for reviewing the compliance with the existing regulations and the YVL safety guides, including current plans for the nuclear waste management.

4.1.3 Periodic safety reviews

In Sweden the operating licences are usually not limited in time. The purpose served by the licence renewals in Finland is, instead, served by periodic safety reviews.

Thus, starting soon after the TMI accident in 1979, the Swedish reactors are, by decision of the Swedish Parliament, subjected to periodic safety reviews once every 8-10 years. The reviews are carried out by the licensees and reported to the SKI for review. The SKI submits the licensee's report to the government together with a report of its own assessment, a so-called ASAR (As Operated Safety Analysis Report).

It should be noted that the ASAR is not a licensing document, but just one documented safety assessment among others - e.g. those obtained in regular inspections and various safety reviews - used by the SKI in its capacity of a safety authority.

The ASAR programme has been concerned, in particular, with management and quality issues, performance records, past and current safety issues and plans for future safety improvements. The efforts were in the early eighties for large part devoted to developing the PSA methodology and undertaking the first plant specific PSA analyses. The reviews of the first and second generations of the Swedish reactors in Sweden have all been completed in the second ASAR round. The first ASAR round is about to be completed for the newest reactors, i.e. Forsmark 3 and Oskarshamn 3.

The design bases and the deterministic safety analyses, as documented in an FSAR, have yet not been extensively reviewed under the ASAR programme, however. For the time being such purpose is served by the special design basis reviews described below (section 4.1.4).

The periodic safety reviews were intended to provide important input in regard of the modernisation needs. The experience has not been entirely positive, however, as can be noted when comparing the improvement programme for the reactor Oskarshamn 1 in the ASAR issued in December 1992 with the improvements which actually turned out to be required in the renovation carried out during the years 1993-1995. An explanation may be seen in that the ASAR programme is in fact not part of a licensing procedure but only part of SKI's inspection and assessment programme.

4.1.4 Special design basis reviews

An event at the Barsebäck plant in 1992 prompted, by revealing inadequate capacity of the strainers in the emergency cooling water systems of the older Swedish BWRs, a campaign with thorough and systematic reviews of all design bases and relevant safety analyses for all Swedish reactors, including the PWRs. The main emphasis was placed on the older reactors. The reviews are estimated to employ several hundred man-years in the industry before completion about in 1998 for the BWRs and somewhat later for the PWRs, due to dependence on obtaining certain design information from the vendor, Westinghouse.

A few issues have, thus far, been brought to light in these reviews, requiring measures to be taken. Thus, for example, a need was identified to adapt the capacity of the emergency boron injection systems at several BWRs to comply with primary coolant leakage rates according to the Technical Specifications for Operation as well as with changing requirements in regard of recent fuel designs.

The reviews aim in the first place at full reconstitution of the FSARs, to ensure compliance with the current conditions for the operating licenses also in the light of new knowledge and modern methods of analysis. Another aim is to assess the safety of the reactors as for possible needs to adapt to the advancing technology, standards and safety practices.

The special design basis reviews are expected to form important bases for the modernisation of the reactors.

The strainer issue led to certain modifications also in the Finnish reactors, which were less affected by the problem, however, than the first two generations of the Swedish BWRs. The confidence in the on-going, continuous reassessment of the licensing bases according to Finnish practice thus remained intact and needs were not seen, as in Sweden, for a broader review of the design bases.

4.1.5 Severe accident analysis

The protection of the environment against severe reactor core accidents is one of the most important aspects of modernisation, as illustrated, e.g., by the European Utility Requirements currently under development.

On account of the accidents in TMI and Chernobyl, the safety regulations in Finland and Sweden call for all reactors to be provided with severe accident mitigation systems for enhanced protection of the environment in the unlikely event of a severe accident. All reactors in Finland and Sweden have been upgraded accordingly.

The severe accident research has nevertheless continued in both countries for further verification of the protection provided by the accident mitigation systems, largely based on participation in international research. Finland and Sweden thus both participate in the Cooperative Severe Accident Research Programme (CSARP) managed by the U.S. NRC, and the EC Phebus Programme, as well as in the Nordic research programme (NKS/RAK-2).

Issues covered in the research, conducted jointly by the safety authorities and the utilities in Finland and Sweden, include, in particular, core melt progression, vessel melt-through and failure, core debris coolability, melt-coolant interactions and steam explosions, and hydrogen burns and explosions. The research aims largely to developing and evaluating the methods and procedures for accident management.

Despite the efforts spent in research, uncertainties still remain, in regard of several of the issues mentioned, requiring continued attention [ref. 21].

4.1.6 Probabilistic safety analysis

Probabilistic safety analysis (PSA) plays an important role in providing bases for the continuous, gradual improvement of the safety of the plants and for the current large-scale modernisation programmes.

PSA is used extensively in Finland and in Sweden for systematic and increasingly detailed evaluation of the safety against the probabilistic safety goals on the basis of deterministically established failure modes and success criteria together with thorough inventory of initiating events. The practical use of PSA in the daily planning of operations, maintenance, plant modifications and operational risk follow-up ("living PSA") is also actively promoted.

The methods used in PSA have been considerably refined and extended in scope through the years, e.g. with regard to "common cause initiators", CCI*, particularly in the electric systems, hazards, like fire, flooding and seismic events, and the risks associated with refuelling and maintenance outages. The scope has also been extended to include, in addition to the core damage risks (PSA Level 1), the corresponding risks for radioactive releases (PSA Level 2).

In Finland, the TVO utility has completed PSA studies of both outage conditions and level 2, and the Loviisa utility is expected to submit corresponding studies to the STUK during the spring 1997.

PSA studies of outage conditions have been reported for four of the Swedish reactors and full scope PSA level 2 studies for five reactors, out of the twelve Swedish reactors in total.

4.2 Safety criteria for modernisation

4.2.1 Safety goals

As pointed out above (section 2.2.1, page 2), the safety goals are set in terms of their scope and the safety level aimed at.

Scope

The measures implemented in the Finnish and the Swedish reactors in order to mitigate the consequences of severe core melt reflect a considerable extension of the safety goals by including consideration of very severe accident conditions beyond traditional design basis.

The quantitative safety goals thus set in regard of severe accidents are primarily aimed for *prevention of land contamination*, which would cause severe social consequences, and for preventing fatalities due to acute radiation disease. They consist in fixing absolute limits to the release of radioactive materials bound to causing severe land contamination, such as iodine and cesium.

In Sweden the limiting release has been fixed at to 0,1% of the total core inventory of such radioactive materials per 1800 MWt. The limit applies to a thermal power rating of 1800 MWt and is to be adjusted accordingly. Severe accident events of extremely low likelihood, such as spontaneous pressure vessel rupture, are not required to be taken into account in providing for meeting the limit.

In Finland the limiting release in regard of severe core damage accidents is fixed at corresponding to 100 TBq of Cs-137.

Recent extensions of the safety goals include accounting for seismic hazards and hazards related to non-power operation of the reactors. In view of the generally increasing safety requirements, there is thus a correspondingly increasing attention paid to the seismic

safety in regard of safety modernisation of the Finnish and Swedish reactors, although the seismic activity experienced in Finland and Sweden is indeed low. The latest generation of the ABB Atom BWRs, Forsmark 3 and Oskarshamn 3, were originally built to seismic standards.

Improvements of the safety during maintenance outages are achieved by more strictly controlled procedures rather than plant modifications.

Probabilistic targets

The total core damage frequency (or rather its inverse), estimated by means of probabilistic safety analysis (PSA, cf. above), is used as a figure of merit of the safety systems* in setting targets for the safety to be achieved. The PSA targets recommended by the IAEA International Nuclear Safety Advisory Group (INSAG) are 10^{-4} or less per reactor year for existing nuclear power plants and 10^{-5} for future plants [ref. 22,23]. The target set by the Swedish utilities is 10^{-5} or less per reactor year for all operating reactors, equal to what is required according to the EUR [ref. 7]. The same target, 10^{-5} per reactor year, applies to the Finnish utilities as required by STUK [ref.24].

The Swedish utilities have adopted a probabilistic target of 10^{-7} per reactor year, or less, of exceeding the prescribed release limit of 0,1%, to be compared with 10^{-5} for existing reactors (INSAG) and 10^{-6} for future reactors according (INSAG and EUR). In Finland, there is a target of 5×10^{-7} or less per reactor year (ref. 24) for exceeding, in a severe accident, a limiting release defined in ref. (16).

4.2.2 Safety approaches

Modern safety principles, intended for future reactors, are applied to existing reactors as can be deemed reasonable in view of the resulting improvement of the safety of the plant as a whole. However, as there are restraints to fully applying novel principles in the older plants, it has to be decided, in practice, if the desired level of safety is indeed conditional on applying the modern safety principles, or could be satisfied otherwise. According to experience this is often the case.

The approach taken follows the recommendations by INSAG in the document IAEA/INSAG-8 [ref 5]:

- checking in the first place, *using knowledge and methods available today*, that all licensing conditions are indeed duly satisfied, accounting for back-fitting²;
- systematic review of the safety of the plant for identification of any deviations from modern safety principles and practices with assessment of their *deterministic safety implications*, again using today's knowledge and methods;
- probabilistic safety analysis (PSA), using state-of-the-art methodology, for assessing the *safety significance* of any found deviations in regard of the internationally recommended probabilistic safety targets (cf. section 4.2.1, page 18).

Following general remarks may be made in regard of progress and approaches taken in modernisations for safety in Finland and Sweden, in regard of the possibilities seen (cf. section 2.4, The scope of safety modernisation of ageing plants):

² In the event that departures would in fact be revealed further operation must obviously be prohibited, or an exemption applied for, until measures to reach compliance have been duly taken.

Prevention of challenging events - the first line of defence

The modernisations comprise for large part modifications aimed at enhancing the quality of the plant processing and control systems. They include, e.g., replacing primary system piping to achieve increased resistance against corrosion and cracking in applying modern materials as well as considerably improved methods and practices for non-destructive testing and control. Numerous modifications have been made relating to the processing systems* and the process control systems, including their various components, as well as the control rooms. Developments of the procedures for reliability centred maintenance and the technical specifications for the various modes of operation also form important parts in the modernisation of the plants.

The safety systems - the second line of defence

According to modern practice, *redundancy**, as applied to the safety systems*, should amount to $n+2$, n being the minimum number of trains required to perform a safety function*, rather than $n+1$ trains required to just meet the single failure criterion*. The main purpose of the stricter requirement is to allow for one train to be out for maintenance during operation at power. The more advanced practice is already implemented in the recent ABB Atom BWRs in Forsmark, Oskarshamn (unit 3) and Olkiluoto, and is being considered for extensive application in the modernisation programmes for the older reactors.

The older reactors feature a great deal of *diversity** in their safety systems*, which partly compensates for their lesser redundancy as compared with the newest ABB Atom BWRs. Common cause failure, CCF*, constitutes in fact, as indicated in safety analyses of the recent generation reactors with greatly enhanced redundancy, a significant limitation to the reliability which can be achieved by extended redundancy. The safety of these reactors will therefore benefit from modernisations in terms of enhanced defences against CCF. Additional diversity constitutes, nevertheless, an important part of the needs for modernisation of the reactors, the newest not excluded.

The consistent *separation* of all safety related systems throughout the whole plant, together with high degree of redundancy, is the most prominent safety feature of the latest generations of the ABB Atom reactors in Forsmark, Oskarshamn and Olkiluoto. Enhanced separation is at the same time one of the important subjects for modernisation of the older reactors.

Inherent stability and "passivity" are, from the outset, typical characteristics of, e.g., the reactivity control of the type of reactors used in Finland and Sweden. Other examples of passive safety devices are accumulators for cooling water delivering on reduced counter-pressure on LOCA*. The benefit of largely extended use of "passive" safety systems is not obvious, however, and must in any case be verified from case to case. The advantages of the active safety systems will then have to be taken into account, like their ability of allowing actuation and active control at any time when required, as well as more forceful actuation in certain applications (comparing, e.g., the cooling achieved using a high capacity coolant pump vs. relying on natural convection).

In pursuing modernisation of the nuclear plants for safety, particular attention is paid to further enhancing the reliability of certain safety functions. One example is the containment isolation valves subject to heavy duty under accident conditions. Measurement of the water level in the reactor vessel under accident conditions is another example.

The accident mitigation systems - the third line of defence

The safety approach applying to the existing reactors, in regard of mitigation of severe accidents beyond design basis, was put at a fairly advanced level already in 1981 by the decision of the Swedish Parliament that all reactors be back-fitted to meet a specified limit to any radioactive releases due to such accidents.

A similar approach was soon taken as well in Finland. It led to following main features having been implemented in the Finnish and the Swedish reactors:

- prepared procedures and facilities, particularly abundant and diverse supplies of cooling water, for protecting the containment in the event of a severe accident and reaching soonest possible a stable state with the core cooled under water cover;
- engineered measures to be provided enabling the containment function to be retained in the event of a core melt accident, in particular during the first 10-15 hours;
- controlled as well as passively actuated relief of overpressure threatening the containment;

Additional approaches, based on extensive research, relate mainly to procedures for accident management, for large part aiming at improved assurance for preventing melt-through of the reactor vessel and keeping the core melt contained in the vessel to be cooled there.

In Finland, the mitigative severe accident management strategy of TVO is built on the same principles as in the Swedish BWRs. The strategy applied for the VVER reactors in Loviisa had to be partly adapted, however, in regard of their particular design characteristics. The main issues include primary system depressurisation, mitigation of energetic events (i.e. hydrogen burns in the containment), retention of the core melt in the reactor vessel, protection of the integrity of the cavity, and control of the pressure in the containment. A world lead was in effect taken by IVO in developing and validating this approach to accident mitigation in a VVER reactor.

The proposed European Utility Requirements, applying to future reactors, go even further in requiring the reactor containments to withstand any conceivable accident conditions without need for filtered venting and by requiring double containment. However, according to analyses, the mitigatory systems implemented in the Finnish and the Swedish reactors are likely to limit the releases of radioactivity causing land contamination as required in the EUR. There are reservations to be made, however, for remaining uncertainties in the analyses (cf. section 4.1.5, page 17).

Completeness of the defence-in-depth

Internal and external hazards have been extensively considered in the modernisation work, both in Sweden and Finland. Seismic standards apply at present to all modifications of the plants, and all plants should meet a common standard in a matter of a few years. The two fourth generation ABB BWRs commissioned in Sweden in 1985 were built to meet such standards.

4.3 Backfitting and modernisation in Finland

4.3.1 Retrospect

The two reactors in Loviisa, of the type VVER-440/213, were supplied by Atomenergo-export, USSR, for essential parts modified according to western type specifications as presented by the Finnish utility, IVO. They were made operational in 1977 (unit 1) and 1980 (unit 2).

The other two reactors, in Olkiluoto, were ordered by the TVO utility from the Swedish vendor ABB Atom (then ASEA-ATOM). Made operational in 1978 (unit 1) and 1980 (unit 2) they belong to the third generation of the ABB Atom BWRs (cf. section 4.3.1).

The reactors have all been subject to numerous modifications and backfits relating to safety through the years they have been in operation [ref 25, 26]. A summary is presented in Appendix 2, indicating the various kinds of modifications classified in the following categories:

- Integrity (1st line of defence)
- Process control (1st line of defence)
- Functional reliability (1st and 2nd lines of defence)
- Severe accident mitigation (3rd line of defence)

4.3.2 Current modernisation projects at Loviisa and Olkiluoto

Both Finnish nuclear power plants are undergoing comprehensive modernisation projects aimed at enhanced safety as well as at an up-rating of the reactor power. The projects thus involve extensive safety reviews and comparisons of the plants with the latest YVL guides. Furthermore, both utilities have performed environmental impact assessments (EIAs), in regard of increasing the power of the plant units, according to a systematic procedure as required by the Ministry of Trade and Industry.

In the safety review of the plants, many international standards were taken into account, including INSAG-8 [ref 5], and the safety analyses have been for great part renewed. The Final Safety Analysis Reports (FSAR) are, accordingly, subject to extensive revision.

Olkiluoto

The modernisation project at the Olkiluoto plant was launched in 1994. The original design margins, accumulated operating experience, and BWR technology were known to offer favourable conditions for a power increase in addition to the increase by 8% already made in 1984. The reactor power will thus be increased by 340 MW_{th} to 2500 MW_{th}, i.e. by 15%. In combination with an increased turbine efficiency this means an increase of the production capacity of Olkiluoto plant by 125 MW per unit. [ref. 27]

The modernisation project is organised in over 30 subprojects. Each subproject consists of an analysis phase, for establishing the design basis, and an implementation phase. The analysis was for main part accomplished in the years 1994 - 1996. The plant modifications are for main part planned for implementation in 1996-1997 at Unit 1 and in 1997-1998 at Unit 2.

In the analysis phase, the current design basis have been compared to up-to-date safety requirements. Based on assessment of the safety significance of deviations found, deci-

sions have been made on modifications to be required. The impact on the safety is analysed deterministically, considering also severe accidents, and by means of PSA. According to the analyses the modifications are expected to entail a significant increase of the safety together with considerably improved productivity of the plant.

The modifications relate mainly to the reactor, the turbines, and the electrical generation systems. They include, furthermore, modernisations of the condensate and the feedwater pumps, renewal of the main transformers and generators, and renewal of some automation systems. The implementations are made in conjunction with the annual outages starting in 1996.

A preliminary version of the final safety analysis report has been submitted to STUK. This will form the basis for a final safety analysis report of the modernisation.

Loviisa

The modernisation project at the Loviisa power plant started in 1995 and is planned to be completed in the year 2000. Together with the other objectives, as mentioned above, the project also aims, in particular, at providing a good basis for extending the plant's life. The project was planned so as to draw as much as possible from modern technology, operating experience, expert knowledge about the ageing processes in performing a thorough reassessment of the safety in the light of the developing safety standards [ref. 28].

A pilot feasibility study concerning the upgrading of power output was carried out in 1994. As a result of this study, the main needs for modification of the plant were identified, as well as necessary additional studies.

The increase of the plant power is to be achieved partly by raising the reactor power and partly by enhanced turbine efficiency. The reactor thermal power will be raised by 125 MW_{th} to 1500 MW. The total electrical output will be increased by around 50 MW per unit. The main modifications relate to the turbine, the electrical generators and the main transformers. During the outage in 1996, the reactor pressure vessel was annealed, which restored the mechanical properties of the steel to nearly original state. Otherwise, only minor modifications in the primary system are necessary. The modernisation of the steam turbines is planned to be completed by the year 2000.

4.4 Backfitting and modernisation in Sweden

4.4.1 Retrospect

The four BWR generations reflect significant developments in regard of the design features related to safety:

1972-76	Generation 1: Oskarshamn 1 Ringhals 1	Safety provisions largely according to the US NRC General Design Criteria (GDC), App. A (draft in 1967, effective in 1971) Complying with at the time unique grace time rule: actions required within 30 mins from alert to be automated
1975-77	Generation 2: Barsebäck 1&2 Oskarshamn 2	Improved separation* and autonomy* of the safety systems*
1980-81	Generation 3: Forsmark 1&2 (Olkiluoto1&2 in Finland)	Typically 4x50% redundant* (n+2: single failure + repair option) and separated trains Internal RPV* recirculation loops* Supports to prevent pipe whip upon breaks
1985	Generation 4: Oskarshamn 3 and Forsmark 3	Further improved layout in regard of separation Seismic design standards

A similar development can be seen in regard of the PWRs on comparing Ringhals 2, made operational in 1975, with Ringhals 3 and 4, connected to the grid in 1981 and 1983, respectively.

In parallel with the stepwise improvements of the safety characteristics of the reactors from one generation to another, all reactors have, since commissioning, been subject throughout to extensive safety improvements by backfitting. As a result, the older reactors have continued to comply with the gradually increasing safety requirements and have in addition been largely kept at par with the newer reactors as far as can be judged by safety analysis, taking prevailing uncertainties into account³.

The past modernisations in regard of safety, disregarding reactive measures on account of certain incidents⁴, can be assigned to following main areas (for details, cf. Appendix 3) [ref 15, 29]:

- Integrity (1st line of defence)
- Process control (1st line of defence)
- Functional reliability (1st and 2nd lines of defence)
- Severe accident mitigation (3rd line of defence)

³ The SKI has accordingly not been in a position to advise the Swedish Government about the order of priority to apply in phasing out the nuclear reactors in Sweden purely in regard of their safety characteristics, as all reactors comply fully with the safety requirements.

⁴ The extensive remedial measures which were undertaken at all Swedish reactors due to weaknesses in the emergency core and containment spray systems as revealed by an incident in Barsebäck on July 28, 1992, thus are not considered here. (Incidentally, similar measures undertaken in other countries, as a result of systematic feed-back of world-wide operating experience and analysis, would on the other hand qualify for being referred to as modernisation.)

4.4.2 The renovation and modernisation of Oskarshamn 1

Although modernisations of the plants in various respects have been going on through the years, the plants of the older generations are presently due for comprehensive modernisation campaigns in order to continue meeting the growing demands in regard of safety and performance. The modernisation of Oskarshamn 1, the oldest nuclear power unit in Sweden, is a forerunner among other campaigns currently in progress. It will be commented in some detail on account of having already reached a fairly advanced stage of progress and presenting a possibly typical case of an ageing reactor in need, in the mid of its useful life, of modernisation as well as a major overhaul.

The design of the Oskarshamn 1 reactor, ordered in 1965, was developed before international safety standards had been established, and at a time when the American rulemaking was only about to become available in drafts. Nevertheless, the design featured already at the outset some of the safety characteristics of today's reactors, such as diverse systems for control rod insertion (hydraulic and electromechanical) and complying with the 30 minutes "grace time rule". A fairly advanced safety feature consisted in an auxiliary condenser, serving as diverse heat sink.

As originally designed, the reliability of the safety systems* was inferior to modern standards, however, providing in particular inadequate defence against "common cause initiators", i.e. initial events implying failure of the safety systems* they require. Although redundancy was adequately provided for, the multiple trains were not adequately protected from, e.g., fire or flooding possibly striking all at the same time. The main emphasis in designing the safety systems, had rather been on defences against major "design basis" accidents, like large breaks LOCAs*. Typically, the improvement programmes consisted for long time for largest part in modifications aimed at ensuring the integrity of the containment structures under such accident conditions.

The Oskarshamn 1 reactor was previously subject, in the period 1979-1980, to one major modernisation in which the power supply system for the reactor was radically redesigned to provide for physical separation of the three-fold supply trains according to then current state-of-the-art. This improvement was shown, by probabilistic safety analysis, to result in the Oskarshamn 1 unit to become largely at par with then newly built reactors.

The renovation project ("Fenix")

The Oskarshamn 1 reactor is presently subject to a comprehensive modernisation programme, scheduled for completion by 1999. The programme commenced in conjunction with a lengthy shut-down period for remedy of problems with inadequate cooling water strainers in the core and containment spray systems of all older BWRs with external circulation pumps, as revealed by an incident in 1992 in Barsebäck 2.

When dealing with the strainer problems, severe cracking was observed in the primary system piping, including feed water piping inside the reactor pressure vessel and some of the vessel internals. This led to a major renovation programme to be undertaken - a project called FENIX - in order to restore the primary system piping to original status and also to thoroughly verify the status of the reactor pressure vessel. The objective of the Fenix project was to qualify the reactor for obtaining permit for restart.

In addition was decided, on having successfully verified the reactor pressure vessel to be remarkably intact after 20 years of operation, to commence a major upgrading of the reactor in regard of safety, a modernisation project.

- Certain modernisation was accomplished already as part of the measures required for permission granted by SKI to recommence operation of the Oskarshamn 1 reactor in the beginning of 1996, such as:
- exchange of all cabling and penetrations in the containment by environmentally qualified components;
- separation of the safety related cabling in four trains within the containment;
- installation of an independent scram function;
- installation of a redundant reference line for monitoring the reactor pressure;
- improved separation of redundant components in safety related equipment by relocation to separate compartments;
- installation of supports for primary piping in locations where possible pipe breaks would be likely to cause damage to safety related equipment and piping.

Certain other modifications, carried out as part of the Fenix project, consisted in remedies of design faults revealed owing to the thorough and systematic analysis required for the safety evaluation of the reactor in conjunction with planning of modernisation programme. Notable deficiencies, which were revealed and remedied, thus included inadequate pressure relief paths in the reactor building, implying threats to its integrity in certain accident conditions - turning them into possible common cause initiators - and remaining, previously unnoticed dependencies in the power supply to the safety systems.

The modernisation project

The modernisation of the plant will be implemented in stages during extended maintenance and refuelling outages in the period 1996-1999. The main objectives of the modernisation and the strategies applied in meeting them, according to a governing OKG AB document [ref. 30], may be summarised as follows:

- The main goal is to provide for durable operation of the reactor for a foreseeable future by meeting, within ample margins, all requirements in regard of safety and performance which are likely to be encountered;
- The goal should be achieved on the basis of consensus, within OKG AB, as to what is required to reach it and to also gain confidence among the public and the authorities;
- The goal should be achieved by means of a robust safety concept, based on modern safety principles applied to the extent possible in an old plant, following the recommendations in IAEA/INSAG-8 (cf. section 4.2.2, page 19);
- The verification of having achieved the goal should rest on a complete account of available experience, completeness in accounting for initiating events and possible event sequences, and verified deterministic analysis for establishing over-all realistic success criteria while accounting for the interplay between man, technical systems and organisation (MTO).

The safety criteria for judging the modernisation needs have been chosen by the utility mainly in accordance with current practice in Sweden. Thus, e.g., the single failure criterion* will be applied while strictly assuming undetectable failures as being present in addition to the single failure.

The probabilistic criteria have been qualified in two respects. Firstly, there is a requirement, for proper balancing of measures applied for safety, that any type of initiating event (with certain categorisation) must not contribute largely to the total core damage frequency (less than 10% when the total core damage frequency is estimated close at the target value, 10^{-5} per reactor year). Moreover, in regard of the uncertainty associated with very low frequencies, as assumed for certain initiating events, e.g. seismic events, the maximum permitted non-availability of any safety function, claimed for meeting the probabilistic safety targets, has been fixed at 10^{-2} per demand. This requirement applies even if the frequency of the initiating event, requiring the safety function, would be estimated, e.g., at 10^{-5} per reactor year or even less, which would otherwise allow for lesser reliability of the safety function.

The modernisation programme consists, for main parts, in:

- replacement of containment isolation valves in accordance with recently enhanced capability requirements;
- replacement of the main steam line isolation valve system to fully comply with the single failure criterion*, i.e. accounting for possible mutual dependency between the internal and the external valves in closing at very high steam flow rates;
- installation of additional supports for high energy piping, as may turn out to be required upon further analysis;
- redesign of the emergency core and containment spray systems, and the residual heat removal system, together with associated power supply equipment, to consistently fulfil established principles for redundancy, independence and separation.
- redesign of the (old type) core spray header and nozzle system to provide for two independent trains of completely separate risers and headers inside the reactor pressure vessel;
- providing for testability of the reactor protection system during operation.;
- further measures in order to eliminate significant dependencies between the process or service systems and the safety systems*;
- further measures for improved fire protection, as may be required according to analysis to be presented.

In addition, renewal of the control room and the equipment for instrumentation and control has been included as part of the modernisation. This part of the modernisation of the Oskarshamn 1 reactor will, however, not be completed until the year 2005.

The successful renovation of the Oskarshamn 1 reactor achieved in the Fenix project indicates, together with the prospects of the on-going modernisation project, considerable potential for extending the life of the older reactors, notably under present market conditions with reasonably short pay-back times.

Approaches and considerations on part of the authority

The SKI followed closely the in-service inspections of the reactor primary system which were undertaken by the utility while taking advantage of the reactor being shut-down for a long period from September 1992 due to the strainer problem. In view of the indications of cracking found in the primary systems, including piping near the reactor pressure vessel, and in the pressure vessel internals, the SKI requested complete verification of the status of the vessel and its internals as another condition for a permit to continue operation of the reactor (the first condition being that the strainer problem would become solved).

The subsequent renovation project - or rather the "start-up project", as lot more than mere renovation was at issue in regard of granting a start-up permit - which started in the summer of 1992 was also closely followed by the SKI. The various safety issues brought to light during the course of this project, as described above, were for main part dealt with separately according to the normal procedure for plant modifications (cf. sections 5.3.2 and 5.4.2). The SKI required in addition, however, for granting a permit to resume operation of the reactor, that the safety case would be thoroughly re-assessed in its entirety, applying, i.a., the IAEA/INSAG-8 recommendations (cf. section 4.2.2).

In order to prepare at the authority for receiving an application for start-up of the reactor, presenting such assessment of the safety case, and to exchange views on the various important issues well in advance, top management meetings were regularly held between the SKI, the SSI (the Radiation Protection Institute) and the utility throughout the course of the project. From the point of view of the authorities these meetings aimed at gaining thorough understanding of the rather complex safety case and offering opportunity of the positions taken on their part to be clarified *in general terms* in regard of information received about the state of the plant, analyses made, and proposed measures. Definite positions in view of specific proposals from the utility were not taken in this process, however, in order to allow for the total concept to be freely judged by the authorities in the end.

The permit granted by the SKI, by the end of 1995, to resume operation of the reactor required it to be decided what modernisations should be made already prior to start-up of the reactor and what should be contained in the modernisation programme to follow. For start-up of the reactor, the principal requirement was *that the safety would be restored, at the very least, at the intended level according to the FSAR, and shown by PSA to meet current probabilistic safety targets*. The modernisation programme to follow, prescribed in the permit essentially as proposed by the utility, aims at complementary safety improvements which would accordingly not be considered urgent but should nevertheless be implemented within the next five years.

4.4.3 Other major modernisation programmes

Despite the recent decision taken by the Swedish Parliament in June 1997, to commence winding up the Swedish nuclear programme by phasing out one of the Barsebäck units (BWR) by 1 July 1998, the utilities plan for continued modernisations of the nuclear plants corresponding to investments of the order of 500 MSEK/year per unit (about \$ 65 M) up to and beyond the year 2000. To give an example, the Forsmark utility will, in the period 1995-2000, be investing 2.000 million SEK (US\$ 300 M) in a "Programme 2000"

for modernisation of their three BWRs, of which two are of the 3rd generation and one of the 4th generation. About 15-20% of the investment is considered to be safety related.

In addition to the modernisation of Oskarshamn 1, described above, modernisation projects are thus in progress also for the other early generation BWRs which were commissioned up to 1977, i.e. Ringhals 1 (1st generation), together with Barsebäck 1 and 2, and Oskarshamn 2 (2nd generation).

Next coming are extensive replacements of piping and joints forming connections to the main recirculation piping in the primary system of Ringhals 1. The project aims at significant improvement of the integrity and the reliability of the barrier function performed by the primary system. The approach includes design modifications to allow improved inspectability and the use of improved materials for the pipe joints to reduce their susceptibility to corrosion cracking (low carbon stainless steel to replace previously used nickel based alloys). The project is expected to considerably reduce the risk for large LOCA* events associated with the use of external circulation pumps* in the 1st and 2nd generation BWRs. Similar projects are expected also for the other BWRs of early generations. The modification is due for implementation during an extended maintenance outage in 1997.

Comprehensive modernisation of the control rooms is planned for the Swedish reactors of all generations, due to commence in 1997 with the third generation ABB Atom BWRs in Forsmark and to be followed by Ringhals 1 and Ringhals 2 (PWR). Modernisation of process control equipment and systems is another commonly planned project, which started by installation of a modern, digital reactor protection and control system in the Ringhals 3 and 4 (PWR) reactors in 1995.

For the BWRs of the 3rd and 4th generations, meeting already in their original design for most part modern safety requirements, there are on-going long range investment programmes aimed at meeting future demands as foreseen for the first decades of the new century.

The needs for safety modernisation will be further clarified in the ongoing special design basis reviews [section 4.1.4] which are due for completion by 1998.

5 THE MANAGEMENT OF PLANT MODIFICATIONS

5.1 Introductory remarks

This chapter is concerned with the implementation part of the modernisation process, i.e. the management of the modification work.

Obviously, safety and quality assurance are prime considerations in regard of modifications of the nuclear plants. Modifications, whether for modernisation or other reasons, thus offer potential for significant safety improvements but also involve exposure of the plant to risks if not properly controlled.

An opportunity developed to study in some particular depth the "process of modification" as it has been implemented in Finland by the Finnish safety authority, STUK, and the Finnish utilities. A rather detailed account of this study is given below. The corresponding Swedish model is described more briefly and comments in regard of notable differences are presented.

A formal modelling technique has been used to describe the procedures involved in the modification process so as to facilitate review for completeness and consistency. The approach taken here, largely in accordance with the so-called Structured Analysis and Design Technique (SADT) [ref 31], proved to be quite helpful, allowing the processes to be broken down in detail as required to account fully for all factors. The technique and the interpretation of the resulting model diagrams are described further in Appendix 1.

There are some problems in being fully clear about all aspects of the handling of plant modifications at nuclear power plants. It is for instance not always clear what is meant by a modification, subject to corresponding procedural requirements. Is a renovation of a system, by replacing its components for fresh ones a modification in the sense that the design basis would have to be reviewed?

It is also not always clear what should in general constitute a single modification, to be dealt with separately for quality assurance and approval by the safety authority. Single modifications may either be combined into larger modification projects and vice versa. Clearly, the safety impact of a modification needs anyhow to be considered in the context of, i.a., other current modifications.

5.2 Regulatory requirements

5.2.1 Finland

Safety rules - the YVL guides

In Finland, all modifications of the nuclear plants concerning components, systems and structures belonging to safety classes 1, 2 and 3 need a permission from the authority. (The safety classification is defined in YVL 2.1). In addition, STUK may request for inspection also other modifications, when considered necessary. For instance, all plant modifications initiated by PSA should be notified to STUK even if the modified part is not safety classified. The main parts of the approval procedure are the review of the documentation supplied by the plant, called the pre-inspection documentation, and the commissioning inspection at the plant. In principle, the phases of supervision for plant modifications are equivalent to those related to the application for construction permit and an operating license.

The basic requirements for the approval of the modifications are given in YVL guide 1.8. Depending on the type and extent of modification, other YVL guides give additional regulations for the approval process. This guide also gives instructions for the approval process of system modifications, for which there are more extensive requirements than for individual component or structural modification. Several other YVL guides give more detailed regulations for the approval process. Figure 5-1 shows how other YVL-guides are related to YVL 1.8.

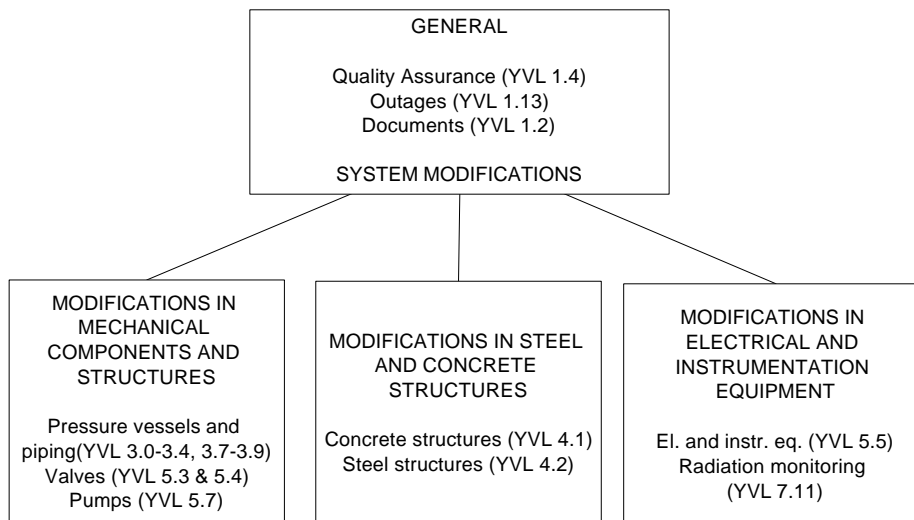


Figure 5-1 Structure of YVL 1.8 and its references to other YVL-guides.

General requirements

In the YVL 1.8 guide, following obligations are imposed on power companies concerning plant modifications:

- the modification plans shall be approved by STUK prior to implementation;
- the power company shall have clearly defined administrative controls and related instructions for the design, implementation and testing of modifications;
- the power company shall use competent personnel in modification works;
- modifications shall be carried out in compliance with the work order/work permit practice;
- there shall be sufficient checks to ensure the resetting to their normal state of process, electrical and instrumentation systems after connections, disconnections and simulations related to modifications;
- the power company shall see to it that requirements approved by STUK as regards radiation, physical and fire protection are complied with in modifications;
- the power company shall have a document updating system;
- the power company shall have systems for the procurement, reception and storage of spare parts.

For modifications in systems belonging to safety classes 1, 2 and 3, a pre-inspection documentation must be sent to STUK for approval. Modification of an individual component is considered as a system modification if this modification significantly changes the operating mode or values of the system.

The pre-inspection documentation should include:

- Causes and justification for the modification;
- System design basis (accounting for safety class, design parameters, ambient conditions, requirements for structural materials);
- Description of the operation of the system's modified part;
- Analysis of the system;
- Any other reports deemed by STUK to be necessary.

5.2.2 Sweden

Regulatory requirements apply in regard of modifications to the nuclear power plants as provided for in the conditions for the operating licenses. A modification is defined as any change to the plant which will require the technical documentation of the plant to be modified.

Traditionally, regulatory approval was required for all "safety related" modifications, i.e. modifications having possible impact on any of the safety features of the plant accounted for in the Final Safety Analysis Report (FSAR). For reasons described above (3.3, page 12), strict regulatory approval is presently required only as expressly provided for in regulations applying in specific areas. Otherwise the regulatory involvement is limited to requiring that statements of all safety related modifications be submitted well in advance to the SKI for decision on which ones will require regulatory review and approval.

Specific requirements apply, e.g., according to regulations concerning mechanical and pressure vessel components which have been issued by the SKI, (SKIFS 1994:1) [ref 32]. These regulations cover components serving for all containment and cooling of nu-

clear fuel and radioactivity and for maintaining the geometrical configuration of the reactor core as required for reactivity control. They do not apply to the fuel elements as such, moving parts in control rod drives or in, e.g., pumps, motors, turbines or generators.

The regulations (SKIFS 1994:1) cover repairs as well as modifications of the plant, in regard of design, manufacture, construction, installation and testing. Manufacturing and construction procedures as well as test methods are required to be specifically qualified for the intended purpose.

5.3 Utility management systems and practices

5.3.1 Finland

The procedure of carrying out plant modifications can be divided into following steps: initiation of modification process, planning phase, implementation and commissioning. Additional important tasks are updating of documents, and training and informing of plant personnel. In following, the main steps of a modification process at TVO are described.

The modification can be initiated either as a result of a development project or by filling up an action proposal. An action proposal is evaluated in the organisation, and a preliminary planning is started after the approval by the manager of the technical unit concerned. A project manager is nominated and a responsible person is selected for the preliminary planning.

The planning at TVO consists of a preliminary planning phase, and planning of the implementation. Appendix 5 shows on a general level the modification initiation and planning phases of TVO. During the preliminary planning, the implementation alternatives are studied taking into account e.g. costs, need of safety analyses and effect on other systems, resulting to a proposal for realisation of implementation planning. The preliminary plan is reviewed and approved.

The implementation planning includes the preparation of all necessary documents, such as drawings, safety analyses, work and testing instructions etc. The documentation of implementation planning is compiled in a modification package. This documentation goes through a review process which considers the effect of the modifications on other systems and plant safety. The modification package is finally approved by the head of operational department. After the approval, the purchasing and preparation of actual work plans start. During implementation planning, the pre-inspection documentation is prepared and sent to STUK for approval.

The implementation of the modification includes installations and testing. The implementation of a safety related modification requires that the pre-inspection material has been approved by STUK - only some preparations can be done in advance. The manager of the technical unit approves the modification on the basis of a test operation report. If the modification needs the regulatory handling, STUK performs an inspection before the commissioning.

The procedures at Loviisa are rather similar, but some differences in the modification planning phase can be pointed out. Appendix 6 shows the initiation and planning phases at Loviisa. It can be seen that in Loviisa there is only one planning phase. Another nota-

ble difference, compared to TVO, is related to the preparation of pre-inspection material. In Loviisa, the preparation of pre-inspection material is an integral part of the modification process, and the plant provides its own guides for the preparation of the documents. At TVO, the pre-inspection material is prepared separately from the plant's internal documentation.

5.3.2 Sweden

Proposals for modifications usually result from well established long range corporate plant development programmes and strategic reactor safety programmes. Continuous feed-back of experience, conducted largely on a joint basis between the Nordic utilities, provides important input.

A plant modification is initiated by the executive of operations for the nuclear plant unit concerned. According to recently developed practice the modification work is managed between the department for operations and other departments, providing technical services, on business-like contractual terms.

Although established in compliance with the general regulations for quality assurance imposed by the SKI, the utility procedures for managing plant modifications exhibit differences in details, reflecting local conditions and preferences. All aim, however, at control of the process on a strict step-by-step basis, from preliminary evaluation of proposed modifications to final implementation in the plant. The general scheme applied to nuclear plant modifications in Sweden may thus be described as consisting of following main *activities*, performed in succession with necessary iterations (cf. Appendix 7):

■ *Preparation of a preliminary proposal*

- registration, assignment of the proposed modification to a review category with regard to safety impact, and designation of bodies required to be referred to for consideration;
- assignment of a responsible co-ordinator;
- preparation of an *evaluation report* covering the backgrounds, objectives, requirements, and a proposed solution as for function, design and design basis together with considered alternatives as evaluated in regard of, i.a., human factors, training needs and cost-benefit;
- preparation of an implementation plan;

Resulting status: Proposal prepared for final consideration

■ *Preparation of final proposal*

- decision by the executive of operations to accept the preliminary proposal for final consideration and to proceed. Issue of an order to prepare a final Modification Proposal ("Ändringsförslag", ÄF);
- referral to assigned bodies for consideration, including for all safety related modifications the quality and safety departments;
- preparation of a *project report* supplemented by a compilation of comments received. The report includes, in addition to the revised contents of the evaluation report, assignment of responsibilities, detailed technical specifications, a func-

tional testing programme and provisions for training. In addition to constituting the basis for deciding on the proposed modification the project report constitutes a *Preliminary Safety Analysis Report* (PSAR), required for approval in case of safety related modifications;

- independent safety review, by the safety department and the corporate safety committee (required for safety related modifications);

Resulting status: Proposal prepared for decision.

■ *Planning*

(usually comprising a number of modifications to be implemented during a maintenance outage)

- decision by the executive for operations to accept the proposal. Issue of a Modification Order ("Ändringsorder", ÄO)
- assignment of a responsible project leader;
- entry into the list of currently planned modifications. This list is subject to review by the SKI for selection of cases for regulatory review and approval in addition to strictly required review and approval according to specific regulations (cf. section 5.2.2);
- final design and documentation of the modification including independent design review against the requirements in the project report and in regard of consistency with the design basis and the functionality of the plant as a whole;
- regulatory review and approval as required (cf. above);

Resulting status: All preparations completed and regulatory approval obtained. The procedural requirements allow actual implementation work in the plant to commence.

■ *Implementation*

- purchasing; construction and installation;
- verification, by inspections and testing under independent third party supervision, that the modification has been implemented as designed and in compliance with current regulations;
- functional testing;
- final documentation, including update of the plant FSAR (Final Safety Analysis Report) and the "STF" (Technical Specifications of Operation) as necessary. In case of major, safety related modifications (e.g. the replacement of the steam generators in the Ringhals PWRs or the renovation of the Oskarshamn 1 reactor) the final documentation of the modification project is required to serve, in effect, as an "FSAR" of the modification project;
- regulatory inspection, review and approval as a condition for resuming operation in all cases involving safety related modifications. In case of other modifications the regulatory involvement is limited to attending meetings regularly

held at the plant for reviews prior to resuming operation, e.g. after maintenance outages;

Resulting status: permission granted to resume operation. In case of major modifications (e.g. replacement of the steam generators in a PWR, power up-ratings etc.) the operating permit may be temporarily restricted to "operation under special regulatory supervision ("Särskild Tillsyn").

- *Operation under special regulatory supervision (when applicable⁵)*
 - verification programme, usually extending over at least one maintenance outage, to be carried out and reported to the SKI;
 - review of the verification report and approval by the SKI.

Resulting status: permission granted for operation under the normal licensing conditions ("tillstånd till kontinuerlig drift")

5.4 Procedures for regulatory approval

5.4.1 Finland

The principal steps in the regulatory approval of a plant modification in Finland are the approval of the pre-inspection documentation and the commissioning inspections of the modified parts. For system modifications, a system test is run after the commissioning inspections of modified parts, and the test results are approved by STUK. Modification of an individual component is considered as a system modification if this modification significantly changes the operating mode or values of the system.

For each modified component or structure subject to regulatory control, a pre-inspection documentation is required according to the corresponding YVL guides. A system modification documentation consists of pre-inspection materials of modified parts of the system and additional system-related documents, e.g. a system analysis.

The approval process of a system modification including electrical and instrumentation components was modelled according to the method presented in Appendix 4. This process model is presented in Appendix 8, and it is further used to compare it with the actual handling process of plant modification in the light of an example in section 5.5.

⁵ *Conditional operating permits*, applying "under special regulatory supervision", are given in circumstances when additional verification is required by the safety authority that the conditions for an unconditional permit are indeed fulfilled. Conditional operating permits are typically granted for the initial operation of a largely modified reactor, e.g. a power up-rating, a replacement of the steam generators or other major renovations, as in case of the Oskarshamn 1 reactor. Such permits are usually associated with special programmes for monitoring the performance during operation and evaluations, as can be made by inspections carried out in a maintenance outage, under supervision of the safety authority. - A conditional operating permit may also be associated with specific authority requirements to be accomplished in certain time, like the permit granted to resume operation of the Oskarshamn 1 reactor in 1996 which is conditional on certain modernisations to be completed according to a time table. An unconditional operating permit may also be declared conditional in case circumstances are revealed which can be expected to require special attention by the safety authority for some considerable time. - According to the policy pursued by the SKI, applications for modifications other than for improved safety are only considered for plants with unconditional operating permit.

System modifications

The pre-inspection documentation of a system modification should include:

- Causes and justification for the modification
- System design basis (accounting for safety class, design parameters, ambient conditions, requirements for structural materials)
- Description of the operation of the system's modified part
- Analysis of the system
- Any other reports deemed necessary

A test run programme shall be submitted for approval together with the pre-inspection documentation, or, well in advance of the system test run. STUK approves the report on the results of the system test.

A system modification is normally composed of modifications of individual components and structures. The processes of accepting a modification of mechanical components and structures, electrical and instrumentation systems and components, and steel and concrete structures are briefly described below.

Modification of electrical and instrumentation systems and equipment

The supervision of electrical and instrumentation systems and components is described in YVL guide 5.5 and 5.2. The procedure described below in inspecting modifications of electrical and instrumentation equipment is based on these guides, although YVL 5.5 will be revised in the near future.

The approval procedure consists of reviewing the pre-inspection documentation, the commissioning inspection and the approval of the performance test. The authority may also supervise the manufacturing and installation of the equipment.

The pre-inspection material consists of following documents: description of the manufacturer, design basis, description of operation, data on type tests and operating experience, and the quality control program.

The draft of the new YVL 5.5 gives requirements for automation systems. There are some additional aspects in the approval of modification of automation systems. E.g. in the pre-inspection documentation, a safety assessment of the automation system is required, and the effect of the modification on the probability of core damage must be presented. For a programmable system, the documentation must include a validation plan according to which the system is validated.

Modification of mechanical components and structures

For modifications of mechanical components and structures, the approval procedure consists of the inspection of the construction plan, the construction inspection, and the commissioning inspection.

The construction plan is the pre-inspection documentation of a mechanical equipment. The contents of the construction plan are defined in the YVL guides (Figure 5-1) related to various mechanical equipment. As an example, the construction plan of a pump, according to YVL 5.7, consists of following documents: description of the manufacturer,

design information, material description, quality control program, sizing of the pump, drawings, information on the motor, auxiliary systems, measuring and control equipment, installation plan of the pump and its motor, information on type tests and operating experience.

The construction inspection is defined in YVL 1.15. According to the safety significance of the equipment and structures, they are divided into three construction inspection classes. For the most safety significant equipment, the inspection is performed by an inspector of STUK or appointed by STUK. In the second class, the inspection is performed by an inspector approved by STUK. In the third case, no inspection is required. The inspection class is defined on the basis of the construction plan documentation. The inspection procedure consists of the inspection of the documents of manufacturing and quality control, the inspection of the equipment, and the loading, tightness and functional tests.

Modification of steel and concrete structures

The inspection of modifications in steel and concrete structures consists of the inspection of the modification plan, and the combined construction and commissioning inspection.

Inspection of the plant modification routines

STUK has developed their own internal guidelines for the inspection of plant modification routines (ref 33). Inspections according to these guidelines were started in 1992. First the inspections were planned to be performed annually, but the schedule has been changed because the need of inspection has decreased as the power plants have improved their reporting practices. During the inspections, the inspector verifies that the plant modifications are carried out according to the requirements of YVL guides. Topics considered in these inspections are e.g. the handling of modifications at the plant, reporting and informing of modifications to STUK, training of plant personnel, updating of documents, and filing of modification planning documents.

The inspection programme has resulted to several improvements in the reporting practices both between STUK and the utilities, and inside STUK. The utilities must inform STUK in advance about planned modifications. This requirement was added in YVL 1.5 guide "Reporting nuclear power plant operation to STUK" revised in 1995. Furthermore, the reports on realised modifications supplied by utilities have improved. STUK has also developed an internal informing practice about ongoing modification works.

5.4.2 Sweden

The involvement of the SKI in the modification processes at the nuclear plants consists partly in general supervision, by on-site inspections in conjunction with the maintenance outages, and partly in direct participation, as described above (section 5.3.2).

In all cases requiring regulatory review and approval of a proposed modification (cf. section 5.2.2), approval by the SKI is required before any manufacturing or construction work in the plant is allowed to commence. Subsequent inspections and testing will then also be required to be approved, as duly certified, for permit to be granted to resume operation of the plant.

Whether regulatory review and approval shall be required, when not strictly called for according to the regulations, is to be decided by the SKI chief inspector in charge of the

plant concerned. In case of doubt the matter shall be referred back to the SKI specialist departments concerned. The criteria and the procedures applying to selecting the cases for regulatory review and approval are currently being revised as part of the process in striving towards focusing the supervision to those modifications which are the most important in regard of safety.

5.5 Performance examples taken from Finland

Two examples from actual practice in the processing of plant modifications are given below. In both cases complications arose as the safety authority, STUK, was not fully satisfied with the documentation and the analyses submitted by the plant. The examples were recommended for this report by STUK. A comparison is made between the actual course of actions in the first example and the prescribed procedure, based on YVL 1.8 and YVL 5.5 guides as described in Appendix 8, in order to identify the differences between guides and practices.

5.5.1 The renovation of main circulation pump frequency converters at Olkiluoto (TVO)

Background

As a part of the plant modernisation program, TVO wanted to replace old frequency converters of the main circulation pumps with modern standard equipment, provided with a digital control system. The replacement of the frequency converters was a prerequisite for increasing the power of the plant.

TVO applied for replacement of the old frequency converters by replacing one converter at unit 1 in the 1996 maintenance outage, four converters at unit 1 and three at unit 2 in 1997 and finally one in unit 1 and three in unit 2 in 1998.

Handling of the modification

A list of events in exchanging documents between the utility and the authority is given in Table 5-1. This list shows the time of each event from the preliminary information given on the planned modification to the approval of the test report. The study was limited to this time period although the complete procedure took longer time. Reference is made in the table to the prescribed procedure for system modifications including electrical and instrumentation components as depicted in Appendix 8.

STUK was not fully satisfied with the documentation submitted, and pointed out, in addition, that the time for reviewing the documentation was too short. The modification was approved with comments and additional requirements, thus allowing the case to proceed. It was required, in particular, that the safety significant programmable systems and their software would be subjected to evaluation by an independent body. Additional information was required concerning the precautions taken in the design of the frequency converters in case of malfunction of the power supplies to the main circulation pumps

Table 5-1: List of the initial exchange of documents between the utility and the authority related to the modification process “The renovation of main circulation pump frequency converters at TVO”.

Date	Sender	Subject/Contents	Reference	Relation to standard procedure*)
26.1.1996	TVO	Renovation of the main circulation pump frequency converters: preliminary information on attempted modifications		P1
21.3.1996	TVO	Main part of the pre-inspection material	26.1./TVO	DOC2.1 DOC2.2
28.3.1996	TVO	Additional documents to the pre-inspection material	21.3./TVO	DOC2.1 DOC2.2
18.4.1996	TVO	Additional documents to the pre-inspection material	21. & 28.3./TVO	DOC2.2
26.4.1996	TVO	Additional documents to the pre-inspection material		DOC2.2
9.5.1996	TVO	Factory test reports		-
21.5.1996	STUK	Approval of the modification with comments and additional requirements	26.1, 21.3., 28.3., 18.4., & 9.5./TVO	DOC2
24.5.1996	TVO	Updated test programme and other documents		DOC3 DOC2.2
12.7.1996	TVO	Additional documents		DOC2.2
15.7.1996	TVO	Test result reports		DOC4.2
21.11.1996	STUK	Approval of test reports with comments and additional requirements	15.7./TVO	P4.4

*) refers to notation in Appendix 8

STUK decided to permit installation of one frequency converter for test use during one year. During the test operation, it was found out that the system did not fulfil all specifications. At the time of the end of the study STUK had thus only approved the reports received thus far with comments and additional requirements.

5.5.2 Ensuring reliability of the emergency heat transfer chain at Loviisa

Background

In connection with the project for modernisation and power increase of the Loviisa power plants new analyses on the operation of the emergency heat transfer chain were performed. According to the analyses the operating temperature of the emergency heat transfer chain could, in case of leakage below the water level of the strainer, exceed set limits in regard of ensuring reliable performance. STUK thus considered the performance margins of equipment to be too small and required plans for resolving the problem.

Handling of the modification

A list of events in exchanging documents between the utility and the authority is given in Table 5-2.

In the letter dated 6.9.1996 STUK requested that the modification plan proposed by the utility would be divided in short term and long term actions. IVO supplied a short term action plan, which was approved. STUK then requested complementary analyses of the containment behaviour at possibly higher strainer water temperature and an account of the capability of the concerned systems to perform under the corresponding conditions. Upon receipt of the requested analyses, STUK approved the temporary modification and required the submission of a long term action plan by the end of 1996.

Table 5-2: List of the correspondence between the utility and the authority related to the modification process “Ensuring of the operation of emergency heat transfer chain at Loviisa”.

Date	Sender	Subject/Contents	Reference
11.7.1996	IVO	Plan for ensuring required performance of emergency heat transfer chain with upgraded reactor power	
19.7.1996	STUK	Request for the proposed plan to account for the rated reactor power and recent analysis results	11.7./IVO
2.9.1996	IVO	Plan for ensuring required performance of the emergency heat transfer chain at rated reactor power	
6.9.1996	STUK	Request for dividing the proposed plan in short term and long term	2.9. /IVO
13.9.1996	IVO	Proposed short term actions for approval	
17.9.1996	STUK	Approval of proposed short term actions with additional requirements	13.9. /IVO
19.9.1996	IVO	Account of “pump displacements” for approval	
19.9.1996	IVO	Account of “performance of pumps with high water temperature”	
20.9.1996	STUK	Approval of the analyses of forces and torques exerted on pumps	18. & 19.9. IVO
23.9.1996	IVO	Account of the additional bracing of piping of pumps	
1.10.1996	STUK	Approval of the construction plan for additional bracing of piping of pumps	23.9. /IVO

5.5.3 Comparison of the actual course of actions with the prescribed procedure

In comparing the actual course of actions in the two cases with the prescribed procedure, it can be noted that this does account for the possibility that major additional requirements may be made in response to proposals received by the authority. In the TVO case was noted, in contrast to the prescribed procedure, that tests are performed although the

documentation is not fully approved. The problem of the prescribed procedure is that it does not consider partly approved documents. TVO applied for the permission to install all frequency converters according to their time schedule, but permission was given only to install one in 1996

The connection between documents being exchanged in the prescribed procedure shown in Appendix 8 and those exchanged in the TVO case study is shown in Table 5-1. The preliminary information on modifications is addressed to P1, planning of modification. Most documents sent by TVO are parts of the pre-inspection material of the system, denoted by DOC2.2. The result reports from factory tests are not addressed in any part of the prescribed procedure. Reason for this is the fact that model contains only those parts of the process that are mandatory. Supervision of manufacturing including the supervision of factory test reports is done only when considered necessary.

The main problems seen in the case studies seem to be related to the planning phase. This should accordingly be analysed in more detail. Appendix 5 and Appendix 6 show on a general level the modification planning phases of Finnish utilities. The most significant differences in the practices are following: The planning process at TVO is clearly two-phased, consisting of a preliminary planning and planning of implementation. At Loviisa, there is only one planning phase. Another notable difference is related to the preparation of pre-inspection material. In Loviisa, this is an integral part of the modification process, and the plant has written their own guides for the preparation of the documents. At TVO, the pre-inspection material is prepared separately from the plant's internal documentation named as "modification package". This may improve the quality of the pre-inspection material.

The documentation sent to STUK meets formal requirements, meaning that the signatures are those required in the plant quality handbook. However, the quality of the contents of the documentation could often be improved, which may reflect problems in the checking and approval practices. It is not enough that the documents are just circulated and duly signed by those concerned in the organisation. Improvement could be obtained by requiring a written evaluation statement of the most important documents.

The problems encountered in handling modifications in a safe and reliable way seem often to be related to too tight time schedules. In some cases, it may be advisable to consider temporary modifications in order to give some more time for finding the best possible solution. The Loviisa case study is an example of a temporary modification being approved by the authority. In the TVO case, the tight time schedule led to failure of getting full approval to allow implementation of the modification in due time.

5.6 Some observations from plant modification processes

Certain components and phases may be identified in ideal processing of plant modifications as being vital in order to ensure good performance and quality. An attempt to identify them was made by conducting interviews at the power utilities as well as at the regulatory bodies. Components found particularly important, together with some typical indicators of their presence in actual practice, have been listed in Appendix 9. Phases commonly afflicted with problems, together with examples of possible problems, are similarly listed Appendix 10. The observations are further commented in the following.

5.6.1 Components of an ideal process

Plant modernisations for safety usually reflect considerable ambition in taking a proactive approach. The ambition may sometimes be lacking, however, perhaps due to most of the attention being paid to the daily operation and problems. Such approach is, however, likely to lead to merely reacting to problems which are in any case encountered, sooner or later.

Systematic planning considering all modifications in their full context will obviously help considerably to ensure necessary consistency. Again, there are cases where the overview seems to have been lacking. If the modifications are treated as isolated entities it is all too easy to overlook important interactions.

The availability of well established formal procedures in the organisation for planning and implementation of the modifications is obviously an important matter, particularly when they are numerous and complex. Insufficient documentation and lack of understanding in the organisation of the safety considerations and the need for formalised procedures are sometimes seen as being the root of problems in achieving desired quality of the modification work.

The competence of the designers is naturally crucial for successful design of any plant modification. Sometimes shortage of competent personnel and heavy workload seem to force the utilities and authorities to allocate inadequate resources for important tasks. This applies especially to reviewing of the safety cases.

A good and confident relationship between the power utility and the authority is a very important factor. The need for careful updating of all documentation affected by the modifications is to be emphasised. Feedback from earlier modifications should also be collected to improve the processing of future modifications.

5.6.2 Phases in the modification process

Various problems can develop when a modification is carried out. Some of the problems seem to have common causes, which are imbedded in the organisation. It is important for all involved in the processing to understand both the basic safety philosophy of the plant and the interactions between various subsystems. Otherwise the modifications may not be well considered and the reviewing for approval of the modifications may also fail in successfully catching mistakes and flaws.

The allocation of adequate resources to the design and review of the modifications is a common problem. Apart from the fact that the organisations are usually kept minimised this is partly due to inadequate planning for contingencies requiring additional work to be pressed into an already tight time schedule. The importance of providing for improvements in this regard is to be emphasised.

As a consequence of the planning problems at the utilities the authorities are commonly put at undue time pressure in reviewing the applications for modifications. The situation is commonly that, at the time the authority receives the application with all detailed documentation, the utility has already selected the manufacturer and contracted services for the implementation and is only waiting for the permission to start installations. This is also usually required in order for the utility to present the required, detailed documentation to the authority. As a result of the time pressure on part of the utility the documentation is also commonly submitted in several pieces, sometimes even including revised

documents to replace documents already submitted; all this naturally causing problems to the authority. Finally, quality problems in regard of the contents of the documentation are likely to arise.

The authorities do not specify times required for their handling of the applications from the utilities. While striving at meeting the demands of the utilities to reasonable extent they are, on the other hand, forced to setting priorities in dealing with the applications and to set policies aimed at ensuring adequate quality of the safety assessments. There is accordingly a tendency towards the authorities requiring that the safety documentation, as submitted, is indeed complete and otherwise to avoid granting partial permits. In particular, it should be avoided to grant partial permits in cases they would imply significant economical commitments on part of the utilities or cannot for other reasons be easily revoked, if found required after complete assessment of the safety case on a complete basis.

Another problem is concerned with the coordination of various activities and keeping track of the current state of commonly numerous modification activities at the same time. While several modifications may be scheduled for a maintenance outage, single modifications may require several outages as well as time during power operation. Good planning tools are thus essential.

The final installation, testing and handing over of a modification to the plant operating department also sometimes are met with problems. Again, these are commonly related to difficulties in creating adequate awareness and understanding of the implications in the entire organisation.

6 SUMMARY, CONCLUSIONS AND RECOMMENDATIONS

The safety practices in Finland and Sweden have been studied and compared in regard of effecting modernisation of the nuclear plants, considering new technology and advancing safety requirements, e.g. as proposed for new reactors.

6.1 Modernisation

Conventions and rulemaking

Finland and Sweden are committed to the Nuclear Safety Convention, adopted in 1994, which requires the contracting parties to review their nuclear plants and to ensure that all reasonably practicable improvements are made to upgrade their safety. Guidance as to what may be considered safe enough is provided by internationally recognised principles (IAEA/INSAG-8) [ref 5] for assessment of the safety of reactors built to earlier standards.

Evolving standards for new reactors, like the EUR [ref 7], and upcoming safety concepts of advanced types obviously influence the notion of what is safe enough in regard of old reactors. These developments are accordingly followed closely, not least in Sweden although no new reactors will be built there. The final safety analysis reports (FSAR) of the operating reactors, forming their licensing basis, are all subject to continuous update to reflect the current safety requirements as they apply to each particular reactor. The Finnish safety authority issues, in addition, a comprehensive set of safety guides establishing the currently required level of safety. These so-called YVL guides thus apply in general to the extent specific exemptions have not been granted by the authority.

Similar, generally applicable safety rules - in addition to the existing safety requirements applying to specific operating licenses - are presently issued also by the Swedish safety authority, given the necessary legal competence in 1993.

Safety targets for modernisation

The safety targets applying in Finland and Sweden, in regard of limiting the probabilities for core damage accidents and severe releases, remain similar and essentially in agreement with previously established international recommendations (IAEA) [ref 4].

Strict limits in regard of quantitative releases in the event of severe accidents apply since the eighties, essentially equivalent to those presently proposed for the European Utility Requirements (EUR) in regard radioactive matter causing land contamination.

Corporate and regulatory goals and strategies

The corporate strategies in regard of modernisation reflect the desire of the utilities to stay in reasonable control of the future of the business by maintaining the plants at highest possible performance/cost ratio and preventing at all times the operation of the plants to be questioned in regard of their safety. Safety is ranked highest among the business objectives by all nuclear utilities in Finland and Sweden.

In Sweden, where phasing out of the nuclear plants in for main part unknown order is foreseen, the only practicable strategy on part of the utilities remains being one of essentially disregarding from this prospect. Modernisation projects are thus typically evaluated at least in a 10-year perspective.

The main strategy of the safety authorities consists in actively promoting continuous improvement of the safety by repeatedly questioning the current safety level in relation to current safety standards and technology and current methods for assessing the safety. The utilities are, furthermore, required by the authorities to consider, at all times, making safety improvements whenever appropriate and reasonably achievable.

According to the "Swedish model", the Swedish safety authority is expected to promote its safety goals so as to first of all allow drawing on the willingness, the competence and the ability offered by the utilities - being responsible for the safety - for reaching the goals. The aims are similar in Finland, but the approaches differ in the degree to which full and detailed verification against prescribed rules is required.

Approaches

The needs for modernisations with regard to safety are explored by the safety authorities in dedicated inspections and safety reviews, in reviewing applications for *relicensing* of the plants, as practised mainly in Finland where licenses are typically granted for ten years, and in *periodic safety reviews* of all plants at intervals of about 8-10 years, as practised in Sweden. The latter practice consists in submitting to the Government a safety analysis report together with a report of SKI's own assessment (ASAR, As operated Safety Analysis Report).

On account of an incident, special comprehensive *design basis reviews* of all reactors are, in addition, under way in Sweden. These aim for large part at reconstitution of the design basis, particularly in regard of the early reactor generations in view of the original documentation being in parts less comprehensive than presently required. The reviews may to certain extent also be seen as reflecting shortcomings of previous periodic safety reviews.

All reviewing of the safety employs internationally well established methodology, covering the technical as well as the human engineering aspects. Probabilistic safety analysis (PSA) based on the extensive feed-back of operating experience made available through the years has contributed largely to the identification of the needs for improvement.

Achievements and current programmes

The four Finnish reactors, commissioned in the period 1977-1982, have all been subject to numerous modifications and backfits relating to safety through the years they have been in operation.

Major modernisation projects are now, since 1994, in addition under way for the two BWRs at Olkiluoto and since 1996 at the two PWRs at Loviisa. The projects aim mainly at verifying the safety, increasing the power rating of the reactors, improving on the thermal efficiency and providing for life extension; all on the basis of recent engineering developments and accumulated experience. They include renewal of a number of main components, mainly in the turbine and electrical generating systems. The modernisation, being implemented during regular maintenance outages, is scheduled for completion by 1998 in Olkiluoto and by the year 2000 in Loviisa.

The safety of the upgraded reactors will be verified in regard of up-to-date safety requirements and in consideration of the evolving safety standards, following, i.a., the procedure according to INSAG-8. The final safety analyses reports (FSAR) will be largely revised based, i.a., on extensive renewal of the transient and accident analyses.

In Sweden, with four generations of BWRs and two of PWRs, all commissioned in the period 1972-1985, considerable modernisations were made from one generation to the other. The reactors of the older generations have, furthermore, been subject throughout their operating life to numerous safety improvements aimed at meeting the safety targets in terms of PSA, which have applied equally to the reactors of all generations.

Nevertheless, the early generations of the Swedish reactors are at present due for more comprehensive modernisations. Also the recent generations of reactors are subject to ongoing, long range investment programmes in order to meet additional demands foreseen for the first decades of the new century. All current modernisation plans still continue to be pursued despite the recent decision taken by the Swedish Parliament, in June 1997, to commence winding up the Swedish nuclear programme by phasing out one of the Barsebäck units (BWR) by 1 July 1998.

The renovation and modernisation of the Oskarshamn 1, the oldest Swedish reactor and essentially a prototype of its kind, is illustrative. It is noted that the thorough analysis required for planning of this modernisation did reveal some previously unnoticed weaknesses despite previous safety analyses, including periodic safety reviews. The approaches adopted for the modernisation of Oskarshamn 1, to be completed in the period 1996-1999, follow closely the recommendations in INSAG-8 (cf. 4.2.2, page 19). Complete reconstitution of the design basis is aimed at so as to meet the safety targets at par with modern reactors, for large part also in compliance with modern design principles.

Recommendations

- for the SKI to continue developing the national nuclear safety rules intended for general application (the so-called SKIFS rules);
- for the SKI to continue - in regard of regulating modernisation of the reactors - establishing safety goals in terms of *documented safety approaches* (cf. section 2.2.1), without being legally binding, to complement the legally binding nuclear safety rules. (Cf. the Finnish YVL guides, section 4.1.1.);
- for the utilities and the safety authorities to actively pursue application of the IAEA/INSAG-8 recommendations for assessing the modernisation needs of the reactors in regard of safety, and to further develop and adapt the recommendations for the conditions prevailing in the Nordic countries;
- for the utilities and the safety authorities to actively follow the evolving safety standards for new reactors, i.a. the development of the EUR, in regard of their implications for assessing the safety of the existing reactors as well as their possible application to them;
- for the safety authorities to assess, in thorough surveys, to what extent the periodic safety reviews conducted in Sweden and the periodic re-licensing practised in Finland serve the purpose of yielding at proper intervals a comprehensive and penetrating view of the safety of the nuclear plants and the needs for further improvements;

- for the nuclear community in the Nordic countries to continue maintaining active and comprehensive safety research programmes aimed at meeting known and unknown demands on a long range perspective.

6.2 The management of plant modifications

The procedures in Finland and Sweden follow largely a common pattern. A modification selected for implementation will typically go through a pre-project phase where a preliminary design for the modification is worked out. The preliminary plan will then be thoroughly reviewed before a final decision to proceed is taken. Approval by the authority is typically required for all clearly safety related modifications. There is a difference, however, in that the Finnish safety authority requires to review and approve, in principle, all safety related modifications while the Swedish authority requires to be presented with all planned modifications for selection of those requiring regulatory review and approval.

The procedures for carrying out changes and modifications of the plants are defined by the utilities in management and quality handbooks, forming parts of the utility quality systems, as required according to general regulations imposed by the safety authorities. Typically, proposals for the modifications are first worked out for approval, after which the detailed design documents are prepared together with procurements plans and test specifications before any changes are made. After implementation, including the necessary training of the plant personnel and revision of the documentation the modification issue may be closed.

The safety authorities strive at playing their part in the modification processes so as not to detract from the incentive on part of the utilities in assuming their full responsibility for the safety. They will also have to consider their own resources, in which respect the conditions in regard of the Swedish authority are the less favourable. In relative terms, the efforts spent on part of the authority, in terms of reviews and inspections on the sites, are thus greater in Finland than in Sweden, covering in Finland also smaller modifications of lesser safety significance, and generally going more in technical detail.

There is, on the other hand, an outspoken strategy of the Swedish safety authority to gain the required assurance of the safety for appreciable part by assessing the quality of the modification process, as being conducted by the utilities, and to lesser extent by taking direct part in double-checking on purely technical matters. Nevertheless, the Swedish authority will usually be heavily involved in significant modification projects including the technical matters.

The procedures practised in making modifications to the nuclear plants appear in general adequate in regard of safety. As far as the practices in Sweden are concerned, there are variations between the utilities, however, which could not be covered in detail in the present study. While expected to be enveloped by the general model presented in the report, their details thus remain to be accounted for. The models displayed for the Finnish practices reflect, on the other hand, accurately the actually prescribed procedures.

Certain weakness could be identified in regard of the rigour of the Swedish practice in selecting the modifications which require regulatory review and approval.

The modification processes have been modelled using a formal modelling technique. The modelling focuses the basic concepts of processing inputs, outputs, controls and resources. The approach aims at facilitating review of the process for completeness and

consistency of applied procedures and allows them to be broken down in detail to account for all factors.

Two examples of recent plant modifications in Loviisa and Olkiluoto have been investigated in greater detail. The examples, as chosen, illustrate the problems related to the utilities working under time pressure. They are considered to reflect a common problem in the processing of plant modifications. Consequences are seen in reduced quality of the documentation submitted to the safety authority resulting in an increased burden placed on the authority.

Recommendations

- for the utilities to consider applying formal modelling of the management and the processing of plant modification, for further improved management control of plant modification work, possibly with the aid of computer;
- for the utilities to further consider the needs for resources in order to cope with the modification work in order to provide additional insurance in regard of quality and safety;
- for the safety authorities to develop practical criteria to be used in selecting those particular modifications proposed by the utilities which should be strictly subjected to safety review by the authority, as opposed to other types of modifications for which approval by the authority may possibly rest on less rigorous quality measures in combination with thorough reviews undertaken at random as judged appropriate from time to time;
- for the safety authorities to evaluate selective approaches to reviewing safety related modifications in regard of cost-effectiveness and impact on the safety culture.

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Appendix 1: Explanation of some terms

Autonomy	Capability of a safety system* or a safety train* to perform its intended function fully on its own, independent of functions provided by other safety systems, and independent of common supply systems.
Barriers	Activity barriers, cf. Defence-in-depth*
CCF, Common cause failure	Simultaneous failure of several safety trains* or components due to a common cause (e.g. a maintenance error affecting all trains). The risk of CCF can be reduced by diversity* and autonomy* of the redundant trains, preventing them from being exposed to common hazards or environmental conditions, and by appropriate maintenance programs.
CCI, Common cause initiator	An initiating event causing failure of safety systems* required on account of the same event, e.g. fire and flooding.
Defence-in-depth	The provision of several levels of protection of the safety, including <ul style="list-style-type: none"> - successive physical <i>activity barriers</i> (fuel matrix, fuel cladding, primary coolant boundary, reactor containment) preventing the release of radioactive material to the environment; - <i>protection of the activity barriers</i> by multiple <i>lines of defence</i> (line 1: the quality and the inherent safety of the plant systems and procedures, line 2: the safety systems* and procedures and line 3: the mitigatory* systems and procedures;
Diversity	Redundancy* provided by different ("diverse") rather than identical means or components. The purpose of diversity is to reduce the risk of CCF* of identical redundant components. A diverse component (e.g. a diverse safety valve) may differ in regard of, e.g., working principles, design, or manufacture.
Internal/External recirculation loops	The older BWRs have "external recirculation loops", using pumps located outside and below the reactor vessel for recirculation of the coolant in the RPV*. The arrangement requires large piping joining the pump inlet and outlet with the reactor vessel and involves, accordingly, a risk for large LOCA*, in the unlikely case of a pipe break. The safety systems of the older BWRs thus need to be designed to cope with such large LOCA. The large piping under the reactor vessel is eliminated in the newer BWRs with "internal recirculation loops", by using internal pumps for recirculation of the coolant in the RPV.
LOCA, Loss of Coolant Accident	Loss of primary reactor coolant, e.g. as a result of a pipe break or safety valve opening with subsequent failure to reclose.
Mitigatory system	System for <i>accident mitigation</i> in case of failure of the safety

systems*.

Appendix 1 (continued)

Passive safety system	Safety system* actuated by built-in natural response to changing conditions in need of control rather than by operator actions or external means (e.g. delivery of cooling water from pressurised accumulators actuated by the reduced external pressure following a LOCA).
Processing system	System providing for a function in the power generation system, e.g. the steam generating system of a PWR (cf. safety system*).
Probabilistic target	A targeted maximum tolerable probability for certain type of accident (e.g. core damage or radioactive release beyond certain limit to the environment) per reactor operating year. The use of probabilistic targets was enabled by the development of probabilistic safety analysis since the late seventies.
Redundancy, redundant	The use of more than the minimum number of equipments (n) to accomplish a certain function (e.g. n+1, or more). If the required capacity is 100% and each piece of equipment provides 25% (n=4), full redundancy is provided by five pieces. - Redundancy is used to meet the single failure criterion*.
RPV	Reactor Pressure Vessel.
Safety function	A specific purpose that must be accomplished to assure safe shutdown of the reactor and lasting heat removal from the reactor core in case of any event or accident condition foreseen in the design.
Safety system	System providing fully or partially for a safety function*.
Safety train	Safety subsystem providing fully or partially for the function of a safety system*.
Single failure criterion	The requirement that a safety function* shall never be hindered due to a random failure of a single component.
Separation	Protection of redundant components and systems from exposure to common hazards (e.g. fire) and to consequential failures of initiating events (e.g. flooding and pipe whips following breaks; missiles) by keeping them separated by distance or, better, by locating them in different compartments with protecting walls.

Appendix 2: Examples of modifications for improvement of the safety in the Finnish reactors [ref 25, 26]

Integrity	<ul style="list-style-type: none"> • Improved structural materials (Olkiluoto); • Measures against consequential damage upon pipe rupture (Loviisa) ; • Measures to alleviate the exposure of the RPV to thermal transients and cycling ; • Replacement of peripheral fuel bundles to reduce exposure of the RPV to neutron radiation (Loviisa); • Improved fire protection by improved alarms, added and improved fire barriers, using materials with increased fire resistance, measures against oil leakages, extended fire-fighting equipment and additional water sources.
Process control	<ul style="list-style-type: none"> • Improvements of monitoring and logic applied to actuation of various safety functions; • Improved and extended control room facilities including user interface to the process computer and critical safety function monitoring systems; • Monitoring of the isotope N¹⁶ in the steam line for detecting primary to secondary system leaks (VVER) (Loviisa); • Replacement of plant process computer (Loviisa).
Functional reliability	<ul style="list-style-type: none"> • Prevention of boron dilution, by controlling, i.a., supply of clean condensate and ensuring that initial dilution is made using borated water (Loviisa); • Modification or replacement of safety related active components for improved performance and reliability; • Environmental qualification of safety related components to better withstand accident conditions. Major project in 1987-92 (Loviisa); • Redundant scram relays outside the room where the reactor protection system equipment is located (Loviisa); • Redundant and diverse feed water line protected from fires in the turbine hall (Loviisa); • Completely autonomous* and separate* standby emergency feed water system on site (Loviisa); • Enhanced separation for protection of redundant safety trains against fires, flooding and consequential damage from initial events, e.g. pipe breaks (Loviisa); • Separate cooling of spaces housing electrical and instrument cabinets for defence against CCF* (Loviisa); • Provisions for ensuring adequate drainage of affected spaces upon flooding (Loviisa); • Added provisions for emergency power supply; • Improved starting system for diesel engines to reduce damage risk at starts (Olkiluoto); • Enhanced recirculation of emergency cooling water in the containment (Loviisa); • Redesigned, considerably enlarged strainers, with nitrogen backflush for clearing, for recirculation of cooling water in the containment, in case of LOCA*, to the emergency cooling system and the containment spray system (Loviisa); • Replacement of mineral wool insulation on the steam lines for reflective metal insulation to reduce blocking of the intake strainers in the wet-well by insulation debris torn off due to LOCA. Provisions for preventing debris from falling into the wet-well, for monitoring strainer blockage and for backflushing of the strainers (Olkiluoto); • Provision for de-icing of the band-screens for purification of intake sea water and preventing clams to block the heat exchangers of the diesel engines (Olkiluoto); • Increased boron solution tanks for added emergency shutdown margin (Olkiluoto).

(continued)

Appendix 2 (continued)

Severe Accident Mitigation	<ul style="list-style-type: none">• Filtered containment venting with provisions for automatic flooding of lower drywell below the RPV* and filling the containment with water above core level (Olkiluoto);• Backup water supply to the containment (Olkiluoto) ;• Hydrogen igniters for controlled hydrogen burns in accident conditions (Loviisa);• Spray system for external cooling of the containment (Loviisa);• Reinforcement of containment structures (Loviisa);• Movable, protective shielding of reactor shaft door against core melt (Loviisa);• New emergency command center on site (Loviisa);• Enhanced and added-on monitoring of accident conditions (Olkiluoto).
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Appendix 3: Examples of modifications for improvement of the safety in the Swedish reactors [ref 15, 29]

Integrity	<ul style="list-style-type: none"> • Added supports, snubbers etc, for steam lines, blow-down piping, feed water lines etc. to prevent consequential damage in the event of breaks; • Reinforcements of containment structures in regard of steam blow-down loads (BWR). - In-service leak testing of the containment (PWR); • Application of advanced non-destructive testing (e.g. remote testing of control drive nozzles) • Added equipment for leak detection under the pressure vessel to ensure that leaks which could lead to breaks will be detected; • Provisions for hydrogen addition to the primary cooling water systems for protection against stress corrosion cracking; • Improved fire protection by added and improved fire barriers, improved alarms, extended fire-fighting equipment and additional water sources.
Process control	<ul style="list-style-type: none"> • Improved and added-on control room facilities. Examples: improved indications of tripped isolation and scram chains; added displays in remote control rooms; • Repeated modernisations of the plant process computers to counter obsolescence and gain potential benefits from the rapid development in this field; • Added applications of advanced process control in several process systems; • Improvements of process control logic applied to various transient conditions (e.g. automatic start-up sequencing, scram and partial scram conditions, selective tripping of faulty turbine set, isolation valve closure etc.). Accounting for previously unforeseen needs, e.g. power instability; • Modernisations of process control equipment and systems, e.g. by introducing digital, software-based systems; • Modified and novel systems and devices for improved or diversified monitoring of various process parameters (e.g. RPV water level, neutron flux). Provisions for additional monitoring needs (e.g. vibration monitoring, power stability monitoring);
Functional reliability	<ul style="list-style-type: none"> • Modifications and replacements of, in particular, containment isolation valves and their actuating devices for enhanced reliability under anticipated, heavy loads. Example: power monitoring of motor operated valves; • Environmental qualification of safety related equipment to ensure reliable performance, e.g., under extreme temperature and humidity conditions caused in LOCA* situations; • Adding redundancy* and diversity*, e.g. provisions for redundant, diverse and autonomous* auxiliary feed water and emergency feed water trains. Adding diversity in highly redundant safety systems*, e.g. pressure relief and automatic depressurisation systems; • Provisions for enhanced separation and independency of safety trains* and their redundant supplies of power etc. to achieve adequate protection against internal and external events and common cause initiators. Example: elimination of dependencies via miniature breakers; • Shutdown margins - enlarged boron tanks;
Severe accident mitigation (Barsebäck 1/2: 1985, remaining 10 units: 1988)	<ul style="list-style-type: none"> • Thermal shielding of containment penetrations; • Back-up water supply to the containment; • Automatic flooding of lower drywell on RPV melt-through (applicable in the latest generation BWRs with dry pedestal area) • Filtered containment venting with provisions for filling the containment with water above core level(type FILTRA for Barsebäck and type FILTRA MVSS for the other units) • Enhanced and added-on instrumentation for monitoring accident conditions.

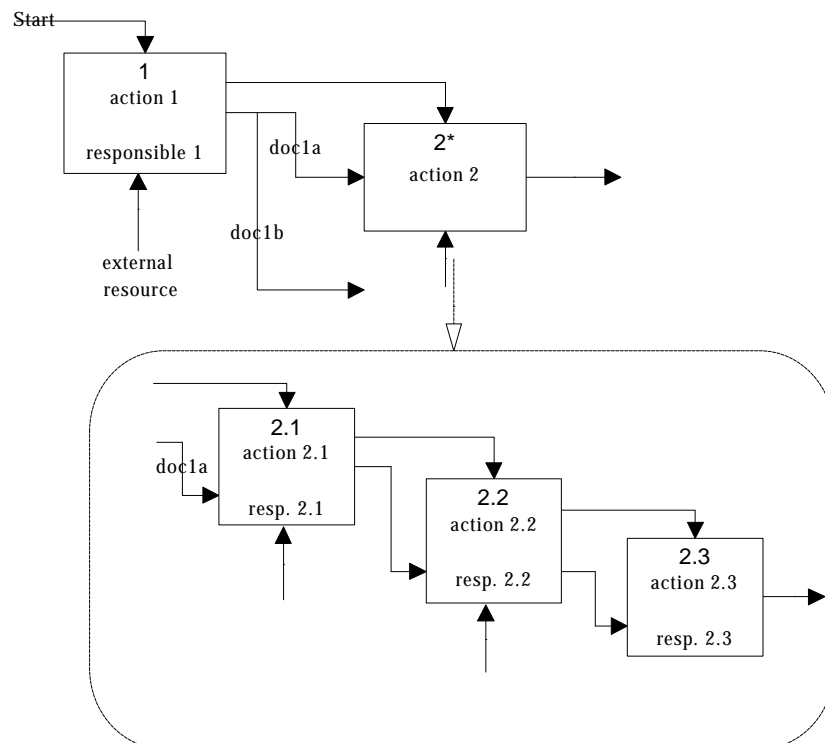
Appendix 4: An approach to analysing modification processes

In a review of the plant modification processes it is advantageous to use formalised models to describe the processes. The formalised model gives a natural language description a structure which makes it easier to compare different processes. They also make it more easy to discuss the actual flow of information through the process, to see the decisions governing the process and to get an impression of the resources which are relied on.

Our plant modification process models are based on a slightly modified version of the Structured Analysis and Design Technique (SADT) [ref 31]. Nevertheless, the basic idea of our modelling technique corresponds to that of SADT: to create hierarchical process descriptions by decomposing large process entities into smaller and more suitable subprocesses (or actions). The process is first described on a more general level, and the subprocesses are progressively broken down until the desired level of detail is achieved. The basic elements of diagrams are boxes, which are used to represent actions in the process. The boxes are connected with arrows. The meaning of an arrow depends on where the arrow is entering or exiting the box.

The notation presented in the adjoining Figure was considered appropriate to our purposes. Each box includes an identification number, the name of the subprocess or action and (optionally) the person or department responsible for the action concerned. The asterisk after the identification number indicates that the action is further decomposed on a more detailed level, as shown for the action 2.

The arrows have the following meanings. The arrow entering the bottom of the box describes



available resources or guides to be taken into account. These resources are considered to be

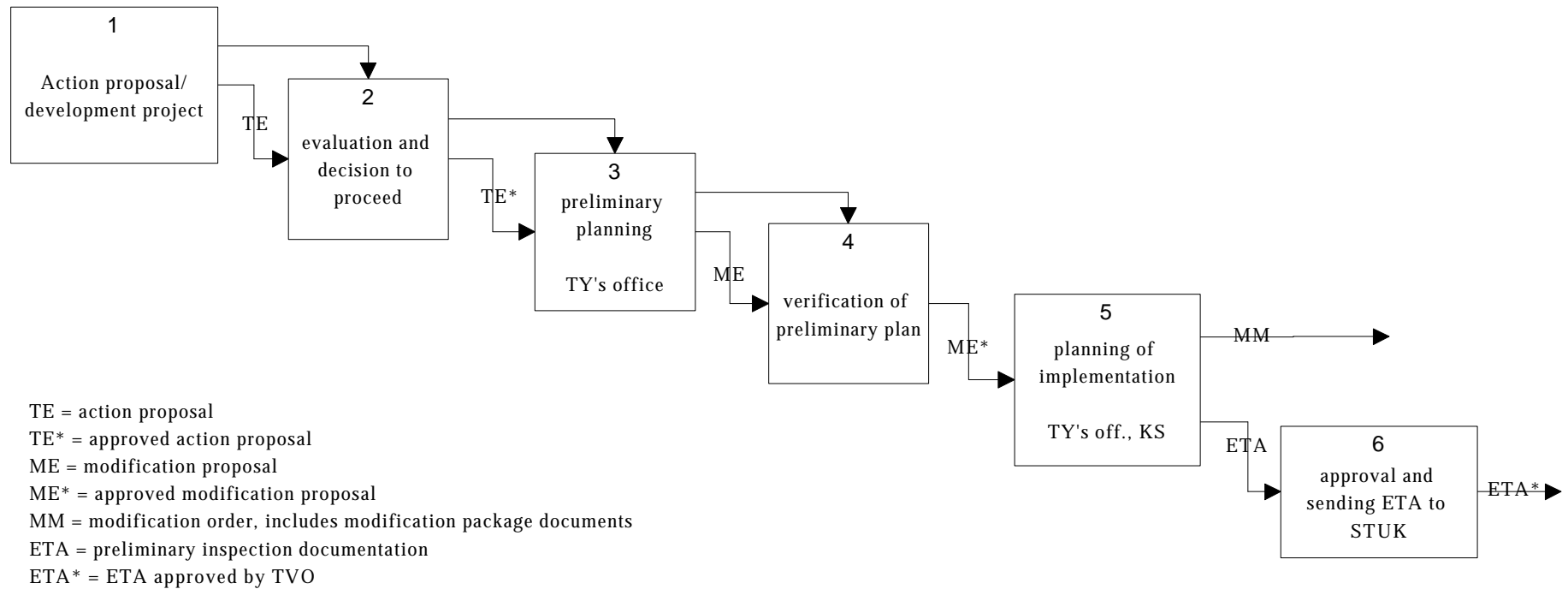
Appendix 4 (continued)

available independent of the process itself or the actual outcome of its predecessor (like YVL-guides provided by the Finnish authorities). The input arrow entering the left side of the box stands for the input from the earlier stages of the process, mainly documentation (doc1a in this example). The control arrow entering the top indicates the prerequisite for the action to be started (e.g. in the figure, the action 1 must be completed prior to starting action 2). The arrows exiting the right side of the box and entering the top of another box represent thus the order of accomplishing the subprojects. Other arrows exiting the right side of the box stand for the outputs of the box, such as produced documentation.

In assessing a described procedure there are some questions to be answered. Where are the processes described, are they unambiguous, can they be easily understood and are they followed in actual practice. Discussions and interviews can give an assessment of how well the processes are functioning in practice.

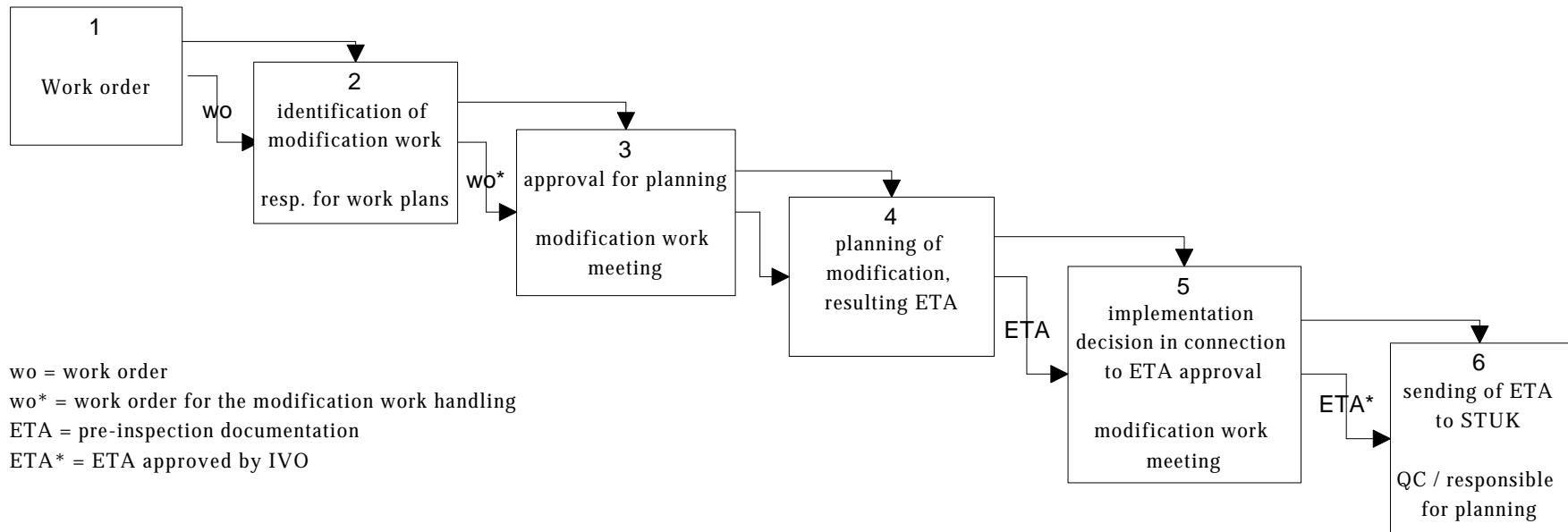
Appendix 5

Modification planning at TVO power plant



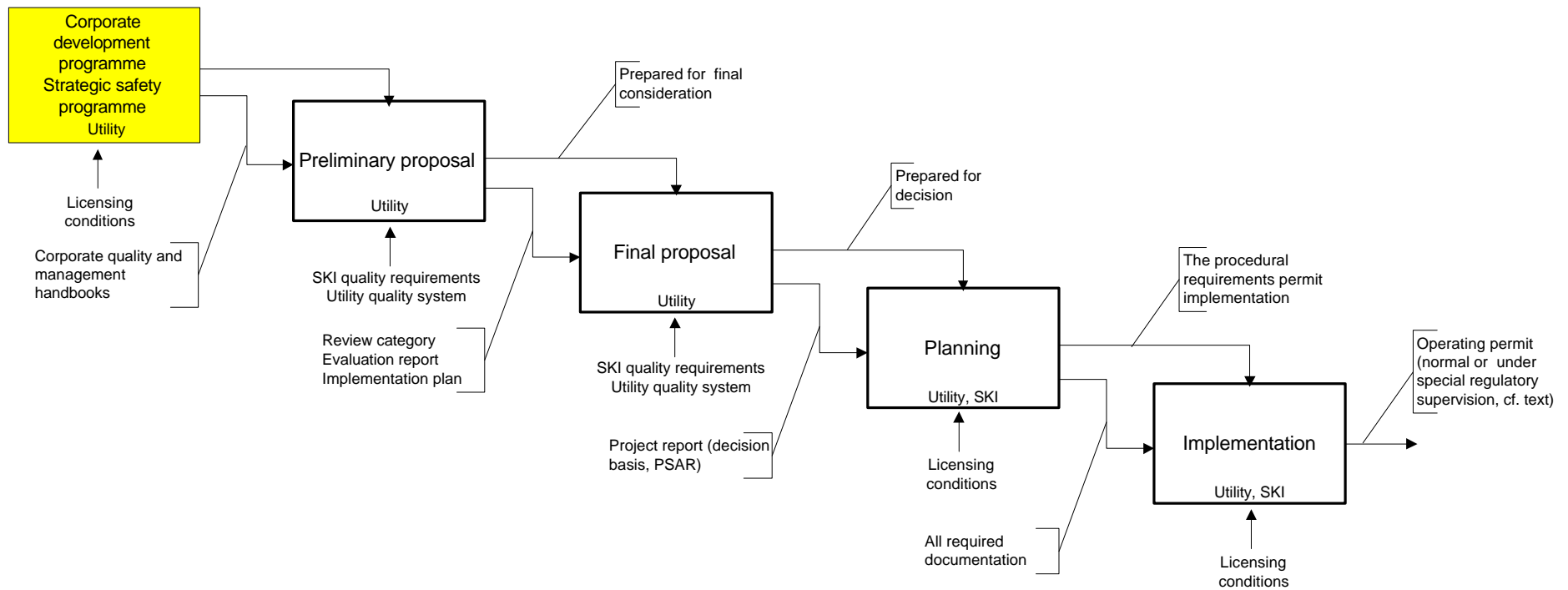
Appendix 6

Modification planning at Loviisa power plant



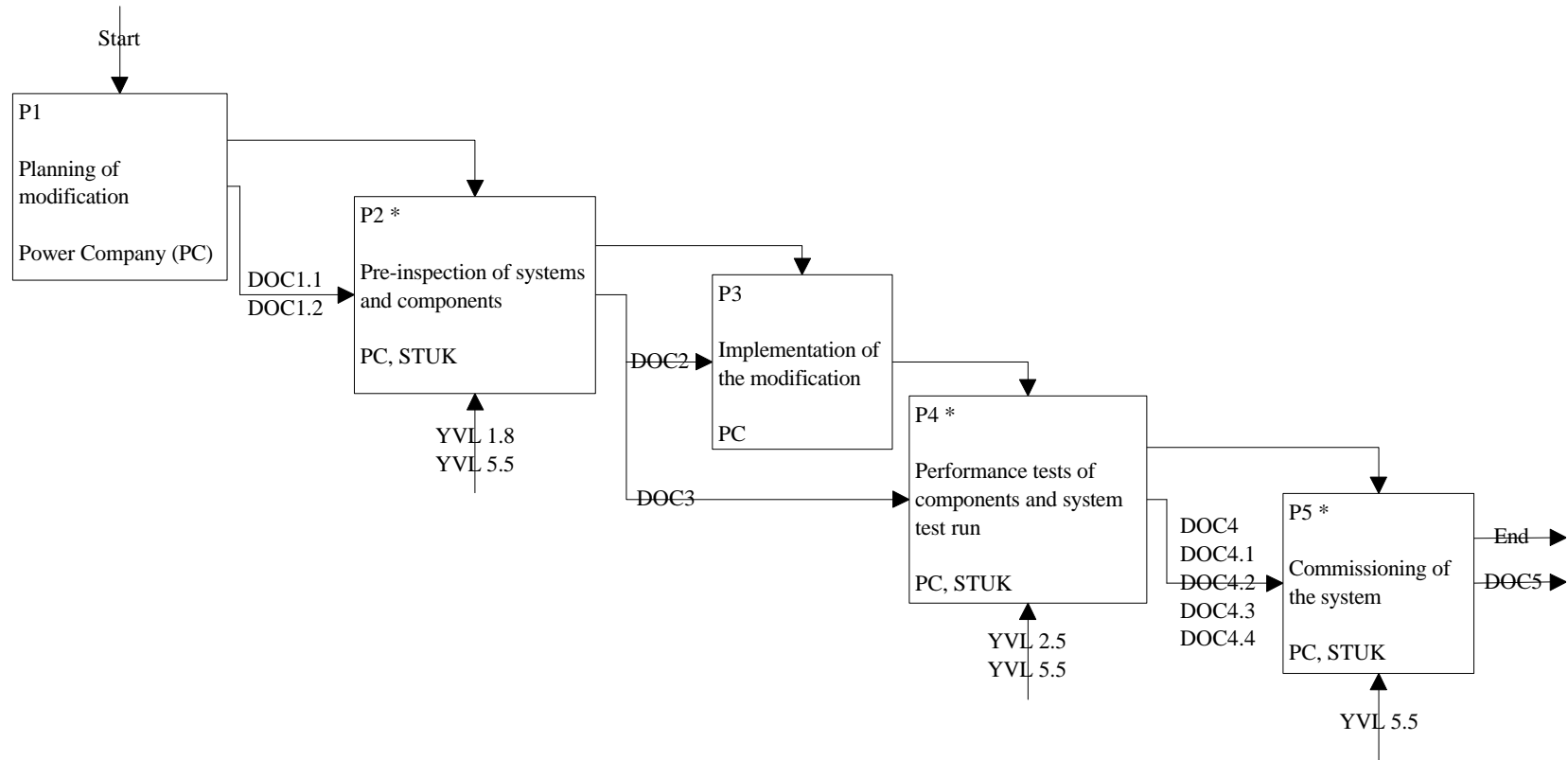
Appendix 7

General procedure for planning and implementation of modifications in the Swedish nuclear plants



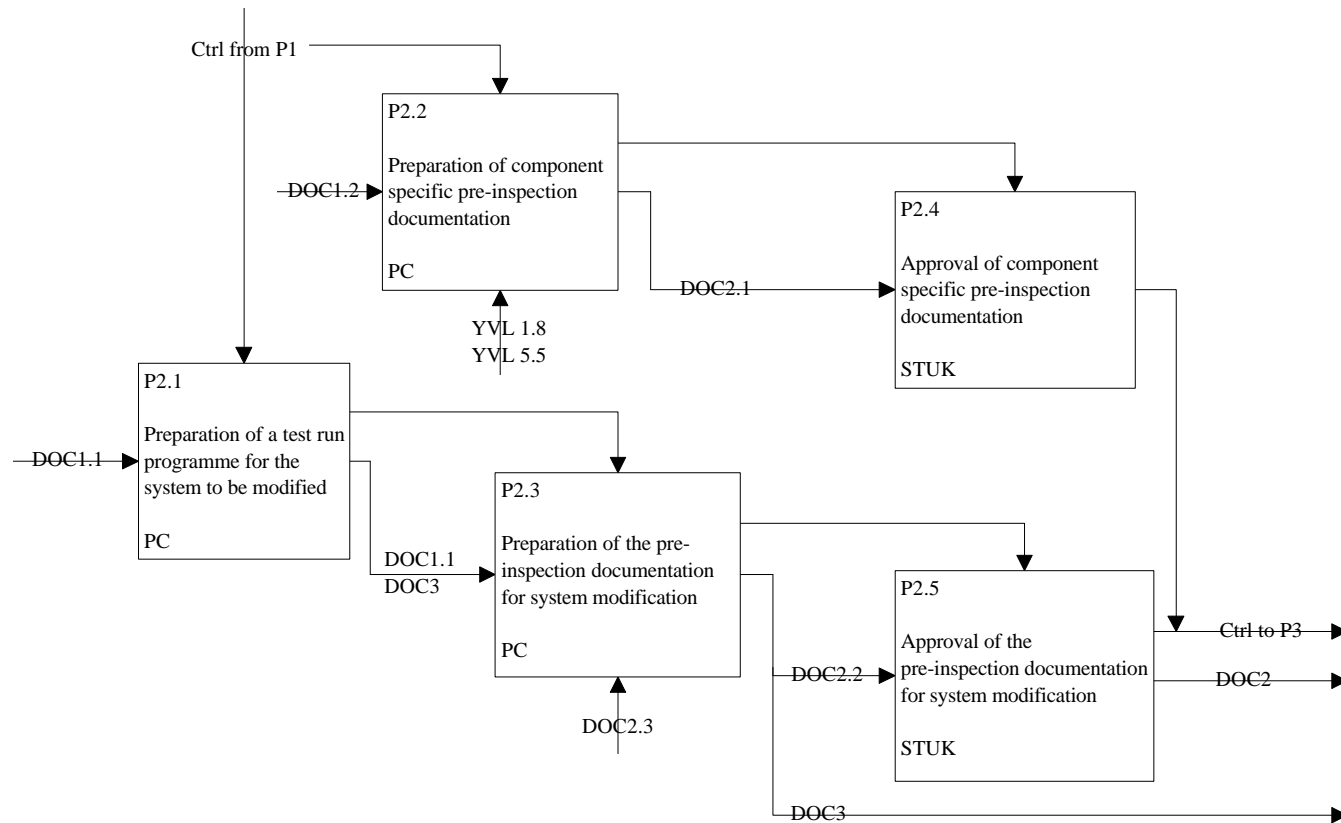
Appendix 8-1: Procedures for modification of nuclear plants according to Finnish YVL Guide 1.8

Modification of system consisting of electrical and instrumentation components



Appendix 8-2: Procedures for modification of nuclear plants according to Finnish YVL Guide 1.8

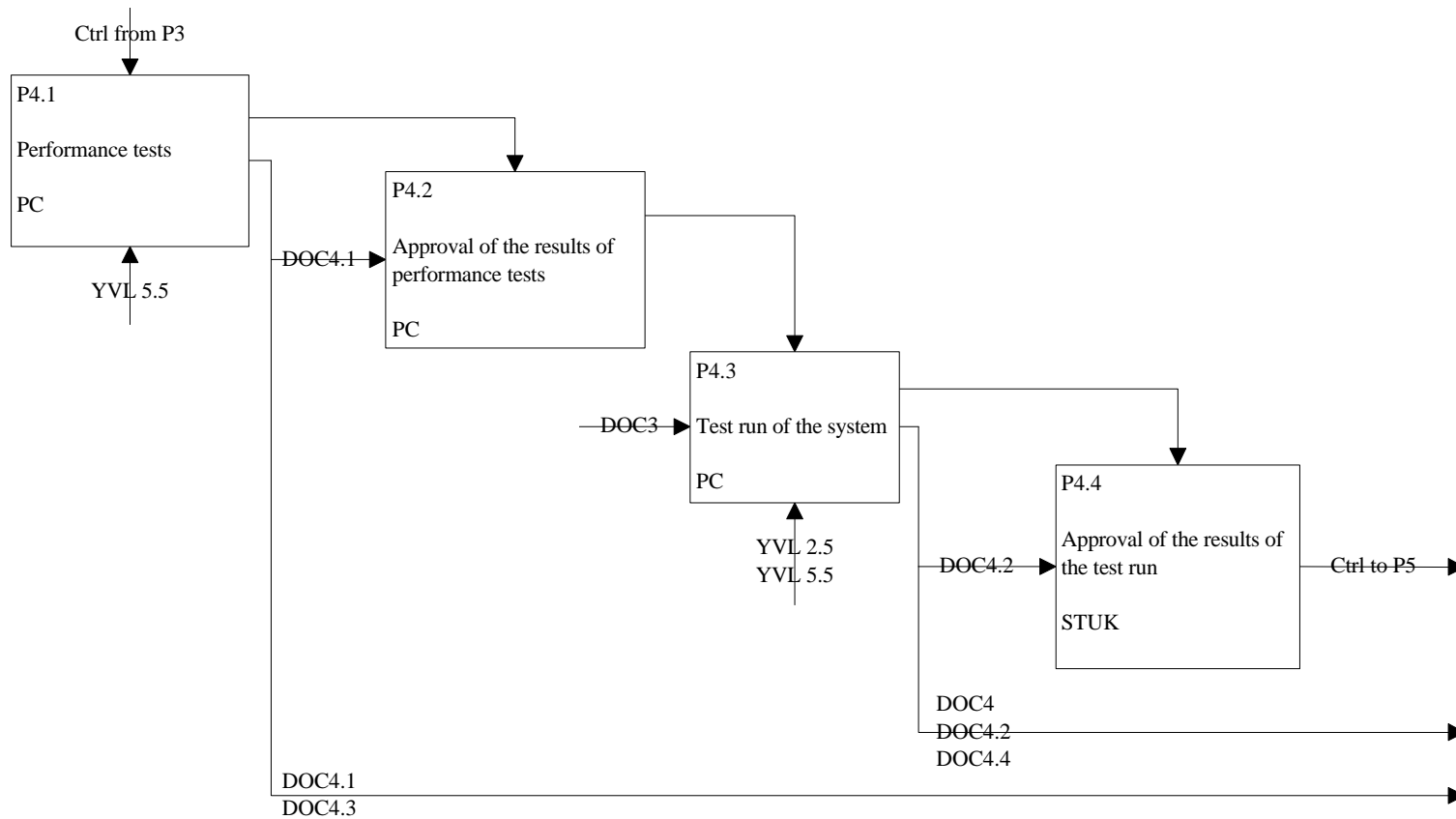
Modification of system consisting of electrical and instrumentation components
P2 Pre-inspection of systems and components



Appendix 8-3: Procedures for modification of nuclear plants according to Finnish YVL Guide 1.8

Modification of system consisting of electrical and instrumentation components

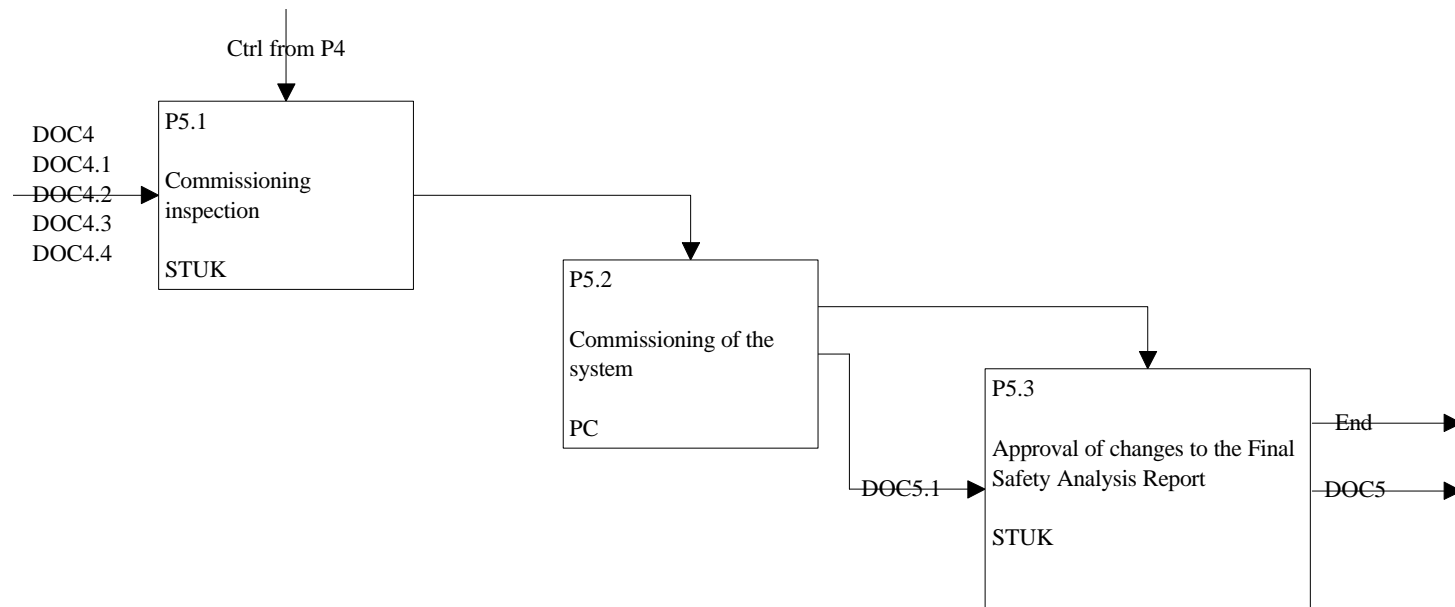
P4 Performance tests of components and system test run



Appendix 8-4: Procedures for modification of nuclear plants according to Finnish YVL Guide 1.8

Modification of system consisting of electrical and instrumentation components

P5 Commissioning of the system



Appendix 8-5: Procedures for modification of nuclear plants according to Finnish YVL Guide 1.8

Information streams

DOC1.1

- Causes and justification for the system modification
- System design bases
- Description of the operation of the system's modified part
- Analysis of the system

DOC1.2

- Description of manufacturer
- Design bases
- Description of the operation
- Data on type test and operating experience
- Quality control program

DOC2

- An approval from the Finnish Centre for Radiation and Nuclear Safety (STUK)

DOC2.1

- Component specific pre-inspection documentation

DOC2.2

- Pre-inspection documentation of the system modification

DOC2.3

- YVL 1.0 and 5.5
- IAEA 50-SG-D3, 50-SG-D7 and 50-SG-D8
- IEEE 279 and 308

- NRC Reg. Guide 1.97

- Fault and impact analysis

DOC3

- Test run programme for the system modification

DOC4

- Application requesting a commissioning inspection

DOC4.1

- Result reports of performance tests

DOC4.2

- Result reports of the test run

DOC4.3

- Results of quality control

DOC4.4

- Response to comments given in the earlier supervisory phases

DOC5

- Final Safety Analysis Report

DOC5.1

- Proposal for changes in the Final Safety Analysis Report

Appendix 9: Plant modification practices - Indicators of good performance

Component of an ideal process	Indicators of good performance
high ambition level	top management policy of placing safety first systematic collection of ideas of improvements follow up of experience and R&D results
quality system	plant modifications are described in reference documents modifications are treated in the strategic planning instructions are available for carrying out modifications
organisational anchoring of modification processes	organisational responsibility is clearly allocated decisions are well documented and communicated persons are committed to their responsibilities
competent planning of modifications	the safety principles of the plant are well understood designers have a long experience in their field the need for a functional entirety is understood
thorough safety review	coverage of a broad spectrum of expertise quality system for ensuring that comments are given routines to ensure that comments are followed up
adequate resources and adequate time allowed	most projects stay within allocated resources time delays are small and not very frequent interviews convey an impression of work without haste
good and confident relationships with the authority	the authority is informed on upcoming modifications disputes on procedures of handling modifications are rare material sent to authorities is comprehensive
updating of documentation and other "clean-up" after the modification	needs for new documentation is assessed clear routines for updating the documentation follow up routines to check that updates have been made
feed-back of experience to improve the processing of future modifications	the follow up aspect is clearly expressed in instructions regular meetings are held at the end of larger projects reports on the modification work are written

Appendix 10: Plant modification practices - Possible problems in various phases

Phase of the modification	Possible problems
initiative in proposing the modification	important initiatives are not carried forward interactions between various systems are not understood initiatives detrimental to the safety are implemented
design of the modification	important couplings between systems are not considered basic safety features of the plant are not understood unsuitable components and materials are selected
review of proposed design	the safety review is not broad enough the safety review is too shallow observed deficiencies are not reacted on
project planning	resources given are insufficient sufficient time is not given lack of competence
implementation	lack of coordination the quality of work is insufficient systems are not made operational after the installation
testing	the test design does not reflect performance under real operating conditions testing is too restricted to exercise all part of the system the results are not interpreted correctly
taking into operation	insufficient checking that the modification has been properly completed and finished, including necessary updates of all concerned technical and safety documents, including training documentation and simulators, as well as other "clean up"