Strålsäkerhetsmyndigheten
Swedish Radiation Safety Authority

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Investigation of long-term safety in the Swedish nuclear power industry and measures owing to the accident at Fukushima Dai-ichi

Summary

Assignment and overall assessment

On 8 April 2010, the Swedish Government assigned the Swedish Radiation Safety Authority (SSM) to, by 31 October 2012, present an analysis of long-term safety in the Swedish nuclear power industry. This assignment encompasses an overall evaluation of the nuclear power reactors' fulfilment of the requirements imposed on safety upgrades, an assessment of which additional requirements on safety improvements that are necessary for extended periods of operation and conditions that may be decisive for operating a reactor over more extended operating periods. The assignment also includes conducting an analysis of the Swedish regulatory model in the field of reactor safety.

Following the accident at the nuclear power plant Fukushima Dai-ichi in Japan, the Swedish Government expanded the assignment on 12 May 2011 to also include presenting a comprehensive report on the results from the new comprehensive risk and safety assessments ('stress tests'), measures taken and planned at Swedish nuclear power reactors as a result of the accident, in addition to conclusions drawn about which additional measures may be necessary at Swedish nuclear facilities.

Following completed analyses and investigations, SSM is of the view that safety at Swedish nuclear power plants can be maintained over the long term as well, provided that additional safety improvements are made and that the licensees apply effective ageing management, and that this is examined regularly in the time ahead in the form of in-depth and periodic safety reviews (PSR). Furthermore, it is essential that a good safety culture is maintained while also ensuring that other safety-related conditions pertaining to organisations and human resources as well as safety-related administration outside the scope of this assignment are maintained and developed in the manner prescribed by legislation and the Authority’s regulations.

Safety improvements to nuclear power reactors

The Act on Nuclear Activities imposes requirements on maintaining safety by taking the measures necessary in order to prevent faults in equipment, incorrectly functioning equipment, incorrect action, sabotage or some other impact that could lead to a radiological accident. This requirement thus implies that a licensee is under an obligation to work continuously on safety and to take measures in pace with the operational experience and as new knowledge becomes available. This is worded more precisely in the Authority’s regulations.

Ever since the Swedish nuclear power plants were commissioned between 1972 and 1985, safety improvements at the facilities have been made when problems have arisen and events have occurred. After the position taken by the Riksdag (Swedish parliament) in 1997 owing to the Government Bill ‘A Sustainable Energy Supply’, which for instance led to removal of the year when the last nuclear power reactor in Sweden was to be shut down, the need for more extensive safety upgrades was accentuated as far as concerns operation over an extended period of time into the future. For this reason, the former Swedish Nuclear Power Inspectorate (SKI) drew up regulations concerning the design and construction of nuclear power reactors, implying that this kind of modernisation work at the nuclear power plants was launched. The regulations (the former SKIFS 2004:2, now SSMFS 2008:17) entered into force on 1 January 2005 with certain transitional provisions. The purpose of these regulations was to give the licensees time to plan and safely perform modernisation work.

Using these regulations and decided transitional action plans as a point of departure, the licensees have since then worked on analyses and measures for fulfilment of the requirements. The transitional action plans originally covered the period 2005 to 2013. This work proved to be much more complicated and time consuming than was foreseen when the licensees produced their proposals and the action plans were decided. Up until 30 June 2012, altogether for the ten reactors’ modernisation programmes, approximately 60 per cent of the decided measures had been implemented. There are major differences between the reactors’ progress, where for a few of the facilities, a great deal of work remains to be done. It is nevertheless important to point out that the measures are of varying safety importance and scope, which is why only comparing the number of measures taken and remaining does not provide an accurate picture of the overall safety improvement nor the progress of the respective action plan. Thus, it will not be possible to conduct an overall assessment of compliance with the regulations on the part of each facility until all the measures have been taken and they have been reviewed by SSM. For this reason, the Authority has intensified its regulatory supervision and follow-ups of licensee work so that remaining measures for fulfilment of the requirements imposed by SSMFS 2008:17 do not take longer than necessary for their safe implementation.

In several cases, the Authority has also drawn the conclusion that the licensees’ safety analysis reports (SAR) on how requirements are applied and verified are too generally worded to give a clear and complete illustration of compliance with the regulations. Reviews have also shown that the licensees’ interpretation of several of the requirements does not agree with SSM’s point of view on the regulations’ implication. SSM is dealing with this area in the ongoing follow-up of the licensees’ work.

SSM assesses that the measures taken and planned to fulfil the requirements imposed by SSMFS 2008:17 strengthen the protective system of the nuclear power reactors' barriers, mainly through increased redundancy and separation, which is the primary purpose of the regulations. Also, when fully implemented, the measures imply a strengthening of the defence in depth system of all facilities. Another safety-related consequence of these measures, other than the purely physical modifications of the facilities, is the improved level of knowledge about the facilities' characteristics that the analyses vis-à-vis the legal requirements of SSMFS 2008:17 have resulted in among the licensees, as well as the fact that the technical documentation concerning the plants has been improved. These conditions are crucial prerequisites for ensuring safe nuclear power plants.

In an international comparison, it appears as if the Swedish nuclear power plants, with these measures taken and planned, are at the forefront in terms of their scope and, to some extent, their progress as far as concerns safety improvements to ageing reactors. In addition to this, Swedish facilities introduced accident mitigation functions at an early stage.

SSM has however in the presently completed analysis of safety improvements drawn the conclusion that further measures will be needed beyond the scope of the licensees' action plans for fulfilment of the requirements of SSMFS 2008:17. The results of the new comprehensive risk and safety assessments (stress tests) also indicate the need for measures in order to strengthen resilience against extreme natural phenomena, a loss of power and a loss of main heat sink. Furthermore, the facilities' emergency preparedness and capability for emergency response management need to be strengthened in various respects. SSM will impose requirements on the licensees to carry out these safety improvements.

The accident at the Fukushima Dai-ichi nuclear power plant has also raised questions requiring more in-depth investigation and research to make it possible to draw conclusions about any additional measures at the nuclear power reactors. These areas are now being discussed in various contexts and research is being planned as part of international co-operation. SSM will take part in many of the investigations and research projects that are launched because of the accident.

SSM has made the assessment that the nuclear power reactors also need to be provided with systems for independent coolant makeup. This kind of system reduces the risk of core melt and thus also the risk of a melt-through of the reactor pressure vessel in the event of a loss of the ordinary coolant makeup system. SSM is now preparing requirements on independent coolant makeup systems, and in this case also intends to impose requirements on the systems' design offering protection which, together with other security measures at the facility, enable the additional coolant makeup system to be maintained to the extent needed in connection with a design basis threat covering the period of time determined by the Authority.

A review of the capability of licensees and the State to protect the facilities against antagonistic threats also indicates that protection against sabotage needs to be strengthened further. Investigations are in progress at SSM to ascertain which additional measures are needed and revisions of the Authority's regulations in the area are under preparation.

Altogether, this implies that the nuclear power plants need to continue their work on analyses and measures at the facilities to fulfil the requirements imposed by SSMFS 2008:17 with decided action plans as well as additional requirements imposed on safety owing to the lessons learned from the Fukushima Dai-ichi accident, stress tests performed, safety investigations and investigations of security. Continued safety improvement measures are also necessary to improve the safety margins against unforeseen events at ageing facilities in long-term operation.

Ageing management and long-term operation

As with many nuclear power plants around the world, when the Swedish nuclear power plants were designed and constructed, a period of operation of approximately 40 years was assumed. This included performing design analyses and fatigue calculations with assumptions concerning a certain number of startups and shutdowns of the plant, other operational conditions, scrams and various kinds of transients during this period of time. Consequently, long-term operation refers to operation beyond the period of time for which the plants were originally designed and analysed. The licensees have announced that they intend to operate the nuclear power plants for 50 to 60 years. Swedish nuclear power plants are presently 27 to 40 years old, counting from the point in time when they began routine operation.

Long-term operation presents both licensees and regulatory authorities with new challenges, starting with the requirements imposed on safe operation during the extended period of operation. The organisation, resources and expertise of both the licensees and SSM must be adaptable to manage new safety issues that might arise in connection with long-term operation. Although many systems and components at the facilities have been replaced over the years in connection with safety upgrades, other refurbishing or rebuilding work and repairs, most critical building structures, systems and components remain in their original design. The licensees' ageing management is for this reason a key area when it comes to safe long-term operation.

In its regulations, SSM imposes requirements on the licensees' ageing management activities in terms of physical and technological ageing, and that an ageing management programme must be in place for this area. An ageing management programme can be viewed as a comprehensive coordination programme consisting of other maintenance and inspection programmes, such as a surveillance programme for reactor pressure vessels, and programmes for environmental qualification, water chemistry follow-ups and monitoring. Requirements are also imposed on the ageing management activities being subject to the licensees' management systems. The aim is to ensure long-term management of ageing issues and as far as possible preventing degradation and other deficiencies from arising so that barriers and components in safety systems no longer work as intended. The requirements imposed on ageing management and ageing management programmes apply generally, but obviously increase in importance as the plants age.

The extensive research conducted nationally and internationally over the past 30 years or so has resulted in increased knowledge about the ageing and degradation mechanisms that can give rise to damage at nuclear power plants. Consequently, these mechanisms can be dealt with satisfactorily with the inspection and ageing management programmes applied today. In turn, these programmes should give good potential for safe operation, also in connection with long-term operation. There are, however, a number of areas in which ongoing inspections and analyses, in addition to development of methods and knowledge, are prerequisites so that these programmes can more effectively detect early indications suggesting safety deficiencies due to ageing over extended periods of operation. These programmes also need to be designed so that they as far as possible are capable of detecting completely unknown damage mechanisms and also known damage mechanisms that manifest themselves in unexpected places.

Also, prior to and during long-term operation, special attention needs to be given to:

- Irradiation embrittlement of reactor pressure vessels, taking account of effects that can substantially increase the rate of embrittlement
- Fatigue, taking account of impact from the reactor water environment on areas sensitive to fatigue
- The condition of tendons and steel liners in reactor containments
- Degradation mechanisms that can influence reactor containments' concrete and metal parts
- Possibilities for reliable inspections and testing of reactor containments
- The validity of environmental qualifications of electrical, instrumentation and control equipment as well as parts with polymer construction materials

Continuous knowledge building is necessary for the long-term application of effective inspection programmes in terms of stress corrosion in: a.) components manufactured of nuclear grade material, and b.) certain materials in pressurised water reactor environments.

As far as concerns even longer periods of operation approaching 60 years, continued investigation and research are needed for timely detection of any deterioration of fracture toughness as a consequence of thermal ageing of stainless steel welds and cast stainless steel.

Thus, what is crucial as to whether a reactor can be operated further over extended periods with a sustained level of safety is the licensee applying a thorough and effective ageing management programme. In the present analysis of ageing issues in connection with long-term operation, SSM has pointed out a large number of measures that need to be taken prior to adopting a position on such operation, while also highlighting the Authority's point of view on necessary parts of the licensees' ageing management activities.

Using this as a point of departure, SSM intends to adopt a standpoint on long-term operation of nuclear power plants on the basis of periodic safety reviews under the requirements imposed by the Act on Nuclear Activities and the Authority's regulations. Clarification and more precise wording of SSM's regulations and general advice concerning periodic safety reviews are being planned, for instance in terms of described aspects of importance for long-term operation. SSM has nevertheless already established that the reporting of a licensee's periodic safety review that is to serve as the basis of the Authority's standpoint on long-term operation needs to encompass analyses describing the plant's ageing status over time for certain key parameters, such as irradiation embrittlement of reactor pressure vessels, component fatigue and tensioning force loss in the reactor containment tendons, for example. This also applies to analyses and when condition monitoring safety-critical electrical cables, instrumentation and control equipment.

SSM's regulation and regulatory supervision in the field of reactor safety

SSM currently has a satisfactory model for regulation and regulatory supervision in the field of reactor safety. This model stands up well in relation to international standards and practice, but it needs to be developed in various respects. This model mainly represents regulation and regulatory supervision focusing on licensees' management and control of their activities and has evolved over the past 20 years, primarily within the regulatory authority of the time, the Swedish Nuclear Power Inspectorate, or SKI. The principles for this model were established in a situation where all Swedish nuclear power plants were to have been shut down by 2010. Among other things, this situation implied that the focus of the regulatory model came to be placed on safety issues related to operation and maintenance. Part of this model includes a general kind of regulation imposing generally worded requirements on the work activities at the Swedish nuclear power plants, also with supervisory work oriented at the licensees' management, control and follow-ups of the organisation's work.

The results of the international peer review (IRRS) performed in February 2012 of nuclear safety and radiation protection work in Sweden and changes in the nuclear industry–such as extensive safety upgrades at the facilities and planned long-term operation plus increased collaboration between regulators and a higher level of harmonisation of nuclear safety supervision–show that the applied model needs to be developed and made more precise. The present analysis shows that this further development and modification needs to encompass not only the regulatory framework, in the form of regulations and general advice, but also regulatory supervision.

SSM's regulatory framework needs to become more comprehensive and be based on international safety standards and European practice, while also demonstrating improved predictability about the implications of the requirements imposed. SSM is now preparing for this revision of its regulations. The challenge for SSM will be to supplement and clarify its regulations and general advice concerning nuclear power safety to enable achievement of these objectives while not going too far in terms of detailed regulation in a way that leads to unclear circumstances about a licensee's safety responsibility. Key points of departure as part of the regulatory framework's revision will be the International Atomic Energy Agency's (IAEA) new safety standards as well as the safety reference levels and other documents agreed between nuclear regulatory authorities in the European Union within the WENRA[1] collaboration.

As up until now, regulatory supervision is to be consistent with the Government's communication to the Riksdag, take note of the controlling element in the supervision while also having an orientation and approach that to a greater extent are adapted to the nature of the various supervisory matters. The objective for the further development and this change is for each situation having a correct orientation of regulatory supervision that is conducted in a way that is fit for purpose and effective.

SSM will consequently, at the same time as the revision of the regulatory framework takes place, develop strategies and approaches for conducting regulatory supervision in the field of reactor safety. In order to achieve a regulatory supervision that is more effective and even more appropriate for purpose, approaches and strategies need to be developed for various areas of regulatory supervision and be adapted to the nature of the areas and matters of regulatory supervision as well as their importance for safety. The starting point for this work will be the findings of the present studies from international co-operation concerning regulatory approaches and strategies.

This means that SSM will be defining the areas of supervision for which different supervisory strategies are to be developed. For each area, it is defined which combinations of approaches are to be applied to serve as the basis for the strategy per area. This strategy is also to be designed so as to facilitate application of the graded approach. In certain areas, consideration will also be given to recommendations from the IRRS review mission as well as recommendations from public sector collaboration as part of the CNRA[2], WENRA and ENSREG[3].

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[1] Western European Nuclear Regulators Association
[2] Committee on Nuclear Regulatory Activities
[3] European Nuclear Safety Regulators Group

This for example applies to periodic safety reviews of the nuclear power plants that are to be conducted at least every ten years. The requirements imposed on these kinds of safety reviews have for a long time now been stipulated in the authority's regulations, but were incorporated in the Act on Nuclear Activities in 2010 on the justification that these reviews are of principal importance. Even internationally, these kinds of periodic safety reviews (PSRs) are of key significance for ensuring that operational experience, new knowledge and new safety standards are taken into consideration and have an impact at the facilities. Against the background of the Fukushima Dai-ichi accident, it may also be anticipated that PSRs will have increasing importance internationally in the regulatory authorities' work to regularly evaluate the validity of the applied design bases, assumptions and safety analyses. SSM will monitor this international development and subsequently adapt the application of the periodic safety reviews thereafter. SSM also intends to give the periodic safety reviews a formal role when it comes to adopting a position on long-term operation of Swedish nuclear power plants in accordance with recommendations from the CNRA and other bodies.

Other areas that have been highlighted by the IAEA, in the international regulatory co-operation work as well as in the present report, and which will be encompassed by SSM's revision of its regulatory model, will include:

- Supervision of ageing management during long-term operation
- Supervision of the licensees' control and quality assurance of work performed by suppliers and contractors
- Analyses and follow-ups of events that have occurred

SSM will also continue development work initiated at the Authority, including annual integrated safety assessments of the nuclear power plants and a more extensive follow-up of safety culture matters.

These changes to the regulatory framework and regulatory supervision mean major challenges for SSM and will require more resources. This applies both to work on changes to the regulatory framework and to the Authority's regulatory supervision, as well as to efforts to examine periodic safety reviews of facilities prior to decision-making on the nuclear power reactors' long-term operation. The IRRS review also drew the conclusion that SSM needs increased resources for meeting the challenges faced by the Authority.

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1 Background

1.1. The assignment

The Swedish Government assigned the Swedish Radiation Safety Authority (SSM), through decision document M2010/2046/Mk dated 8 April 2010, to by 31 October 2012 present the following:

1. An overall evaluation of the nuclear power reactors' fulfilment of safety upgrade requirements imposed by the Authority in regulation SSMFS 2008:17 and the Authority's assessment as to how this modernisation work has had an impact on reactor safety
2. An analysis of the preconditions for operating the reactors over extended periods of time (more than 50 years) in addition to the further requirements imposed on safety improvements ensuing from these kinds of extended periods of operation and developments in technology and science
3. An assessment of which main conditions will be decisive as to whether a reactor can continue to be operated over extended periods while maintaining safety
4. An analysis of the Swedish regulatory model in the field of reactor safety in relation to international standards
5. International experience gained from safety improvements to reactors as a basis for decisions concerning extended periods of operation

On 5 July 2010, the Director General approved a directive for an investigation into long-term safety in the Swedish nuclear power industry (decision document ML 71/2010). On 2 November 2010, the Director General subsequently approved a project plan for this work (decision document ML 88/2010).

On 12 May 2011, the Government expanded the scope of the assignment through a supplement (M2011/1946/Ke) against the background of the accident at the Fukushima Dai-ichi nuclear power plant in Japan. The additional assignment implies that SSM must also by 31 October 2012:

1. Submit a comprehensive report on the stress tests of the relevant Swedish nuclear facilities that were to be conducted in 2011 on the basis of the European Union-wide guidelines,
2. Account for the measures taken by the industry by this point in time owing to the tests and the Authority's assessment of these measures, and
3. Present an evaluation of: a) the areas identified in the stress tests requiring more in-depth analysis, and b) other lessons learned from the accident at Fukushima Dai-ichi, in addition to conclusions drawn on any further measures that should be taken at Swedish nuclear facilities.

Part of the additional assignment involved submitting an interim report by 15 December. The project and investigation plans were updated because of the additional assignment.

Consequently, both of these Government assignments encompass an analysis and investigation in the following three main areas:

1. An analysis of and improvements to safety at the ageing reactors based on new knowledge and safety developments
2. Operation for a period of time exceeding the originally analysed/designed period of operation with a particular focus on surveillance of ageing aspects and ageing management in connection with long-term operation

3. The Authority's regulatory supervision of the maintenance and development of safety in the field of reactor safety

1.2. Approach

This investigation has been conducted in the form of a project with three sub-projects, corresponding to the three main areas described above. This work has followed the updated investigation plan approved by the Director General on 5 September 2011 ('ML' decision document SSM2011-99-61).

Sub-project 1 set up the work on the basis of the following six aspects:

- An updated reconciliation of the requirements imposed by SSM's regulations, SSMFS 2008:17, concerning the design and construction of nuclear power reactors in relation to the requirements applying to the modern Generation III reactors that are now being planned or under construction in different parts of the world
- A survey of safety upgrades made at reactors abroad as input for extending periods of operation in Sweden
- An overall evaluation of each nuclear power plant's progress in its work on fulfilling the requirements of SSMFS 2008:17, including both a qualitative analysis of how the modernisation work has had an impact on safety as well as a more quantitative analysis using probabilistic safety analyses (PSAs)
- A compilation of information on the results of the stress tests performed by the relevant Swedish nuclear facilities owing to the accident at Fukushima Dai-ichi, in addition to an assessment of measures taken by the industry due to the tests
- A compilation and an evaluation of areas identified from the stress tests of Swedish reactors and areas identified from stress tests of reactors abroad and which have a bearing on Swedish facilities. This aspect also includes compiling and evaluating other lessons learned from the accident at Fukushima Dai-ichi that might require more in-depth study in the form of investigation and research
- An integrated analysis and overall evaluation of additional measures that might need to be taken at the relevant Swedish nuclear facilities in order to further improve safety

This integrated analysis and overall evaluation serves as the basis of the conclusions drawn and the recommendations provided in the report in response to items 1, 2 and 5 of the Government assignment as per M2010/2046/Mk 2010, in addition to items 2 and 3 of the additional Government assignment as per M2011/1946/Ke.

The work of Sub-project 2 was based on a previous investigation [1] at SSM concerning regulatory supervision and the preconditions for ageing nuclear power plants' long-term operation. This investigation delved into the following areas from the perspective of long-term operation:

- Irradiation embrittlement of reactor vessel materials
- Low-cycle fatigue considering the reactor water environment
- Thermal ageing of cast stainless steel and welds
- Concrete constructions and reactor containments
- Inspections and testing
- Electrical systems and control equipment

After this stage, Sub-project 2 encompassed in-depth investigations with the aim of:

- Describing and analysing ageing aspects of the assumptions for operating Swedish reactors over extended periods of time

- Assessing which ageing-related conditions will be decisive as to whether a reactor can continue to be operated over extended periods while maintaining safety
- Proposing improvements to the Authority's regulatory supervision and regulations while taking into account the long-term operation of ageing nuclear power plants

Sub-project 2 also included further detailed investigations into how SSM's work on examining the periodic safety reviews under the Act on Nuclear Activities (1984:3) needs to be developed further considering the ageing aspects of long-term operation.

Based on these efforts, conclusions have been drawn concerning important issues related to ageing in connection with long-term operation and in response to items 2 and 3 of the Government assignment as per M2010/2046/Mk 2010.

The work of Sub-project 3 consisted of three areas in stages involving:

- An analysis of the results from the IRRS[4] self-assessment and the IRRS review having a bearing on SSM's regulatory supervision of nuclear power plants
- An updated survey of the supervisory situation to date at a number of SSM's sister authorities internationally
- An integrated analysis and evaluation of the advantages and disadvantages of different regulatory models for supervision of nuclear power plant safety

This integrated analysis and overall evaluation serves as the basis of the conclusions drawn and the recommendations provided in the report in response to item 4 of the Government assignment as per M2010/2046/Mk 2010.

This project was run under the leadership of Lars Skånberg. Anders Hallman led Sub-project 1, Björn Brickstad led Sub-project 2 and Anna Franzén led Sub-project 3.

The following persons also took part in this work: Lars Bennemo, Peter Ekström, Wiktor Frid, Tomas Jelinek, Gustaf Löwenhielm, Fritz Maier, Åsa Rydén, Richard Sundberg, Lovisa Wallin and Kostas Xanthopoulos.

Interim reports [2] [3][4] were drawn up for each of the three sub-projects and serve as input for the information provided by this report.

1.3. Delimitations

This report covers measures to maintain and develop safety at nuclear power plants in particular, but in certain respects the central interim storage facility for spent nuclear fuel (Clab) as well. The investigation excluded:

- Aspects related to radiation protection of workers at the nuclear power plants
- Areas relating to management of low, intermediate and high level waste from the nuclear power plants
- Releases to the environment during normal operation
- Decommissioning matters
- Non-proliferation control and non-proliferation matters

[4] The IAEA assists member states with (for instance) independent peer reviews of agency structure, legislation and public authority work in the fields of nuclear, radiation, waste and transport safety. This is the Integrated Regulatory Review Service, or IRRS, that is performed against the IAEA's standards that to varying extents are relevant to regulatory authorities and public sector work.

The investigation also excluded reviews of the industry's stress tests of nuclear power plants and Clab, as well as producing a Swedish national report for the European Nuclear Safety Regulators Group (ENSREG) and the European Commission. This was done in the form of a special project. An interim report [5] was submitted to the Government on 19 December 2011 and the Swedish national report [6] was presented to ENSREG on 29 December 2011.

Areas relating to measures for the maintenance and improvement of the facilities' physical protection (security) are only touched upon when such measures also have a bearing on an improvement of safety. SSM has for instance dealt with areas related to physical protection within the framework of the Government assignment (M2011/3091/Mk) concerning a review of the capability to protect nuclear facilities and shipments of nuclear material against antagonistic threats.

2. Analysis of improvements to safety at ageing reactors based on new knowledge and safety developments

2.1. Starting points

The Act on Nuclear Activities (1984:3; 'KTL') imposes requirements on maintaining safety by taking the measures necessary in order to prevent faults in equipment, incorrectly functioning equipment, incorrect action, sabotage or some other impact that could lead to a radiological accident. Thus, this requirement implies that a licensee is under an obligation to work continually on safety and to take measures in pace with operational experience and as new knowledge becomes available. This is worded more specifically in SSM's regulations, and has been a requirement since 2009 by means of the Nuclear Safety Directive (2009/71/EURATOM) stipulating that the Member States of the European Union must maintain their respective national framework and improve it as necessary while taking into account operating experience, insights gained from safety analyses of nuclear facilities in operation, technological progress as well as findings from safety research, when these are available and relevant.

The reactors in Sweden were designed and constructed in the 1960s and 1970s and commissioned between 1972 and 1985. When it came to design, construction and commissioning, there were no generally applicable Swedish regulations nor other rules. For this reason, a great deal of the legal requirements in force at the time in the United States came to be applied in Sweden by means of standpoints and decisions taken back then in connection with review work performed by the regulatory authority.

Since then, the basic principle for safety upgrades of the reactors has been gradual improvement through plant modifications and special measures taken as problems arise and when serious incidents take place at facilities in Sweden or abroad. For instance, following the accident at the Three Mile Island nuclear power plant in the United States in 1979, the Swedish Government appointed a reactor safety inquiry whose report [7] included the recommendation that Swedish reactors should be equipped with pressure-relief filters to limit discharges. These filters were then introduced between 1985 and 1989. Another example is the 'strainer incident' at Barsebäck 2 in 1992 that led to extensive rebuilding of the facilities' emergency core cooling systems.

The position of the Riksdag in 1997 owing to the Government Bill 'A Sustainable Energy Supply' for example resulted in the members of parliament supporting the Government's proposal concerning the closure of the two nuclear power reactors at Barsebäck and not setting a specific year for shutting down the last nuclear power reactor in Sweden. Among other things, this situation accentuated the need for safety upgrades at the other facilities for operation over an extended period of time from this point on.

When it came to the oldest reactor, Oskarshamn 1, extensive modernisation work had commenced in 1995 because of certain safety problems and the regulatory authority launched work on defining the requirements that were to apply to operating this and the other reactors during the 2000s.

This work originated from domestic and foreign operational experience, safety analyses, findings from research and development projects as well as from the International Atomic Energy Agency's (IAEA) safety standards. These efforts culminated in new regulations, with transitional provisions, concerning the design and construction of nuclear power reactors. These regulations were resolved by the former Swedish Nuclear Power Inspectorate, or SKI, on 7 October 2004, were issued as SKIFS 2004:2 and entered into force on 1 January 2005. The purpose of the transitional provisions was to give the licensees time for planning and safe implementation of the modernisation work. These regulations are now incorporated in the present Swedish Radiation Safety Authority's Regulatory Code and are issued as SSMFS 2008:17.

Using these regulations and decided transitional action plans as a point of departure, the licensees have since this time worked on analyses and measures for fulfilment of the requirements.

On 11 March 2011, the Tohoku region in north Honshu, Japan, suffered a severe earthquake with an ensuing tsunami. The impact was devastating for the area, including destroyed towns and a very high death toll. At the Fukushima Dai-ichi nuclear power plant, with its six reactors, offsite power was lost first, followed by the facility's own reserve power. When the power supply could not be restored, there was no potential for cooling the nuclear fuel in the reactors and fuel pools, culminating in the fuel superheating and becoming damaged in several of the reactors. In addition to this, hydrogen gas formed, resulting in the need to relieve the reactor containments of pressure into the environment plus leakage to reactor buildings. This was followed by explosions that destroyed reactor buildings, and as a consequence, there was an external release of radioactive materials.

The accident at the Fukushima Dai-ichi nuclear power plant demonstrated the need to once again conduct an analysis and evaluation of safety at nuclear power plants. In Europe, the regulatory authorities imposed requirements on new comprehensive risk and safety assessments of resilience against earthquakes, flooding and extreme weather conditions, in addition to prolonged power failures and loss of heat sink, regardless of cause. Part of these stress tests included analysing and evaluating the facilities' emergency preparedness and pre-planned measures for managing severe accident sequences assuming extensive destruction of surrounding infrastructure as well as radioactive contamination at the site.

In the investigative report [2], SSM has documented measures taken as a result of the Authority's modernisation regulations, how these measures are assessed as fulfilling the requirements and having their intended safety impact, in addition to the need for additional measures. The report also contains a summary account of the stress tests carried out against the background of the accident at Fukushima Dai-ichi, measures taken and planned, as well as the Authority's point of view on additional efforts owing to this accident. This summary is provided in sections 2.2 to 2.11 below.

2.2. Requirements imposed on safety upgrades of Swedish reactors

When the regulations (SKIFS 2004:2) concerning the design and construction of nuclear power reactors entered into force on 1 January 2005, certain aspects implied more stringent and broader requirements being imposed in comparison with the previous regulations. In other respects, the regulations laid down pre-existing principles that applied when designing the reactors, also in connection with subsequent plant modifications. The latter did not imply more stringent regulation in relation to the

requirements applied, but through the general regulations, the requirements were attributed with uniform application and greater transparency while also enabling straightforward communication with various interested parties.
The requirements that were made more stringent through SKIFS 2004:2, presently SSMFS 2008:17, predominantly apply to resilience in different respects against internal and external events. In a nutshell, these can be summarised as follows:

- Automatic or passive functions should be used for safety systems. If this is neither possible nor reasonable, pre-planned manual actions may be accepted if there is sufficient time for consideration[5]
- Operating systems must not have a negative impact on safety systems
- The reactor containment must have been designed while taking into consideration the phenomena that may arise in connection with severe accidents (core melt)
- It must be possible during all events to keep a core, or core melt, covered with water to sustain cooling also over a long-term sequence
- Failure on the part of an individual component is to be managed by the facility applying redundancy[6]
- Multiple failures are to be prevented by means of diversification[7] and administrative measures
- Physical and functional separation between redundancies is to apply for the prevention of the same failure or event from having an impact on several redundancies
- The nuclear power reactor must be capable of managing loads that might arise in connection with pipe ruptures
- The nuclear power reactor must be capable of withstanding natural phenomena and other events outside or inside the plant
- The nuclear power reactor must normally be controlled and monitored from the central control room in all operational states. If the central control room is unavailable, it must be possible to bring the reactor to hot standby and monitor it from the emergency control post
- A reactor core must be designed with sufficient safety margins ruling out swings in output; if output swings do occur, these must be discovered and reduced

2.3. Measures taken and remaining for meeting modernisation requirements

2.3.1 General information about measures taken

As the analyses and measures required for fulfilling regulation SKIFS 2004:2 in certain cases implied extensive efforts, the licensees were, in connection with the provisions entering into force, given the opportunity of assessing the time needed for taking the measures. Using the licensees' assessments as a platform, SKI determined when particular analyses and measures were to have been performed on the part of each separate nuclear power reactor. These action plans originally covered the period 2005 to 2013.

This work, however, proved to be much more complicated and time consuming than was foreseen when the licensees produced their proposals and when the action plans were

[5] Time for consideration = period of time for thinking things through before a decision or a particular action is taken
[6] Redundancy = several autonomous backup systems that can perform the same task
[7] Diversification = a redundancy that performs the same task but in a way that differs in principle or differs in nature

decided. The licensees have also mentioned that their work taking the various measures identified a need for certain additional and to some extent other measures for meeting the requirements. Furthermore, they have stated that taking the various measures within the parameters of modernisation work has been so wide-ranging that it has not been possible to take all the measures by the specified period of time while also maintaining high standards on quality.

Taking these wide-ranging measures also imposes great demands on planning and coordination of various efforts. In most cases, this involves implementing qualified design modifications. The availability of the expertise needed for these measures is limited.

Altogether, these problems have led to some of the measures applied for by the licensees receiving approval for postponement of the deadlines when the respective measure was to have been taken (and consequently when compliance is to have been achieved). The longest postponement of the deadlines for measures is until 2015.

Up until 30 June 2012, approximately 60 per cent of the resolved measures had been implemented altogether at the reactors. The conditions in this respect for each separate reactor are described in SSM's memorandum [8]. It is nevertheless important to point out that the measures are of varying safety importance and scope, which is why only comparing the number of measures taken and remaining does not provide an accurate picture of the overall safety improvement nor progress of the respective action plan.

The measures in question are extensive and many of them are taken to contribute to several of the properties stipulated by the regulations. Moreover, the reactors vary to some extent in terms of their design, also meaning that measures having the same aim differ from one reactor to another. Taken together, this situation makes it impossible for the present report to give a complete account of these measures.

A summary account of the most significant measures taken or planned is presented in the report [2] and memorandum [8]. The report covers the relevant reactors and the deadlines for taking the remaining measures. Brief descriptions are provided below of the measures taken or planned on the part of the respective reactor. These measures largely match those of the decided action plans, though certain additions to these plans have been made. It is also worth mentioning that some modernisation work was launched for reasons other than those necessary for fulfilling the Authority's regulations. For instance, there are cases of instruments and control equipment being replaced due to difficulties in obtaining spare parts, also called 'technological ageing' (see also Chapter 3).

2.3.2. Brief information about the measures taken at the respective plants

Forsmark 1 and Forsmark 2

Forsmark 1 and 2 were commissioned in the early 1980s and were originally designed with redundant systems as part of many safety functions, which were separated to a limited extent. Modernisation work has been undertaken for some safety functions in order to fulfil new requirements on separation and diversification, etc. Measures have been taken to enable cooling over a long-term sequence following core damage. In order to strengthen the protection against common cause failures, initiation of the control rods' screw insertion function has been rebuilt into a safety function. Automatisation of the boron injection has been implemented and additions have been made to initiate the reactor's safety circuits by means of two different parameters, although some parameters remain to be implemented. In order to improve physical and functional separation, the residual heat removal systems have been separated into four trains. The 110 V and 220 V voltage supply has been rebuilt so that operation and safety functions are separated. At Forsmark 2, separation of different redundant trains has been improved in certain relay and apparatus rooms by implementing new fire compartments, and even a fire mitigation system in the form of a reduced oxygen environment has been installed but not yet commissioned. This measure remains to be taken at Forsmark 1. Cooling system buildings have been reinforced and fire sprinklers have been supplemented in cable culverts for the purpose of strengthening protection against internal and external events. Back-up control rooms have been implemented to make it possible to control the reactor in a situation where the standard control room must be evacuated. Measures have been adopted for detection and automatic control of core instability.

Forsmark 3

Forsmark 3 was commissioned in 1985 and, together with Oskarshamn 3, are the most modern nuclear power plants in Sweden that were also designed with separated and redundant safety functions. The modernisation work undertaken has largely been completed for the purpose of fulfilling the requirement on protection against common cause failures, such as automatic shutdown by means of boron injection, an external water source for emergency core cooling and the implementation of functions for diversified residual heat removal. Apart from these measures, the back-up control room has been supplemented and detection and automatic control of core instability have been implemented. Measures remain to be taken for initiating the reactor's safety circuits through two different parameters, also enabling cooling in the long-term sequence following a core melt.

Oskarshamn 1

At Oskarshamn 1, Sweden's oldest nuclear power plant that began routine operation in 1972, deficiencies in its design were identified already in the 1990s. As a result, extensive measures were taken in the early 2000s. Consequently, when SKIFS 2004:2/SSMFS 2008:17 entered into force, OKG assessed that the measures were sufficient for fulfilment of large parts of these regulations. In order to fully meet the requirements, Oskarshamn 1 will be taking further measures. This includes installation of an automatic boron injection system as well as analyses of design basis values for natural phenomena and other external events. Some of the measures pointed out in the transitional decisions remain to be taken.

Oskarshamn 2

Oskarshamn 2 is also one of the oldest reactors in Sweden. It began routine operation in 1975. OKG Aktiebolag has run a project called 'PLEX' for many years now, among other things to analyse, supplement and upgrade the facility so that it will fulfil the requirements imposed by SSMFS 2008:17. Some of the measures that are already finished include installation of a system for emergency venting of the reactor pressure vessel, as well as better separation of the power supply of the auxiliary feed water pumps implemented before the regulations entered into force. Most of the planned measures for fulfilment of the requirements of SSMFS 2008:17 nevertheless remain to be done. Measures particularly worth mentioning include diversification of the reactor protection system, automation of the boron injection system, installation of water discharge valves inside the pressure relief system, diversification of residual heat removal in addition to improved redundancy and separation of the emergency diesels.

However, this project is delayed and OKG Aktiebolag has applied for an extension meaning that the measures would not need to be taken until 2014. SSM has not yet dealt with this application.

Oskarshamn 3

Oskarshamn 3 was commissioned in 1985 and, together with Forsmark 3, are the most modern nuclear power plants in Sweden that were also designed with separated and redundant safety functions. The modernisation work undertaken has largely been completed for the purpose of fulfilling the requirement on protection against common cause failures, such as automatic shutdown by means of boron injection, new logic circuits for water discharge valves, an external water source for emergency core cooling and the implementation of diversified residual heat removal. Apart from these measures, the scram system has been modified to perform its function without pressure relief. Measures remain to be taken for initiating the reactor's safety circuits through two different parameters and protection of the reactor containment during severe accidents.

Ringhals

Ringhals has applied for and been granted permission to make the most extensive changes to the measures stipulated in the transitional decisions. This is largely a consequence of new findings during work on the large-scale modernisation programmes. During the preparations for the modernisation work, the need to both broaden some measures and implement new measures has been notified to SSM.

Ringhals 1

Ringhals 1 is one of the oldest reactors in Sweden. It began routine operation in 1976 and has consequently undergone several large-scale modernisation projects launched before SKIFS 2004:2/SSMFS 2008:17 entered into force. The purpose of one of these projects was to upgrade the reactor protection system and residual heat removal function. In many ways, this project has been part of the original action plan of Ringhals 1 for handling conditions that might arise in the event of design basis accidents. The operation of the nuclear power reactor and monitoring of the barriers' surveillance were also improved by implementing new control equipment (including the modernised reactor protection system). Other measures taken or planned strengthen the barriers and protective functions of the barriers by means of increased redundancy, separation and resilience against impact from fire, earthquakes, lightning strikes and other external events.

Ringhals 2

Ringhals 2 is also one of the oldest reactors in Sweden. It began routine operation in 1975 and is the oldest pressurised water reactor in the country, which has justified the very extensive modifications of control equipment design in recent years. This was done in the form of the 'TWICE' project, which has now been completed. Measures as part of this project were predominantly pre-existing measures from the original action plans. The measures taken at the facility due to SSMFS 2008:17 contribute in several ways to a stronger defence in depth. Operation of the facility and monitoring of the barriers are being improved by implementing new control equipment; here, particular consideration is given to the man-machine interface. Other measures taken or planned strengthen the barriers and protective functions of the barriers by means of increased redundancy, separation and resilience against impact from earthquakes, fire and hydrogen release. Major contributions to the safety improvements are represented by the installation of passive hydrogen recombiners, replacement of control equipment, the planned improvement to the auxiliary feed water system, the planned improvement to redundancy and the improvement to the component cooling system. The qualification of the pressurizer’s pressure relief valves for water discharge is also particularly worth mentioning.

Ringhals 3 and 4
Ringhals 3 and 4 are reactors whose original design is relatively modern. These reactors began routine operation in 1981 and 1983, respectively. Despite this, however, the number of measures in the original action plans for these plants were numerous, albeit of a somewhat different nature compared with Ringhals 2, for example. The measures have to a greater extent been supplementary analyses whose aim is to verify the present design, but to some extent also identify measures that need to be taken for fulfilment of the requirements. These measures include passive hydrogen recombiners and a new fire extinguishing system. Beyond these, it is particularly worth mentioning diversification of the reactor protection system and qualification of the pressurizer’s pressure relief valves for discharging water as measures that will improve safety.

2.4. Fulfilment of requirements and impact on safety

2.4.1. The action plans’ adequacy in terms of compliance

SSM’s overall assessment is that the measures taken to date owing to the requirements imposed by SSMFS 2008:17 have improved safety at the ten nuclear power reactors in Sweden. The main capability that has been improved is their control over conditions that might arise in the event of design basis accidents. The operation of the nuclear power reactors and monitoring of the barriers’ surveillance have also been substantially improved by implementing new or upgraded control equipment.

Viewed in relation to all the facilities, more than half of the analysis and modernisation work has now been completed as per the decided action plans. This means that many measures remain to be taken, particularly at a few of the facilities. SSM’s regulatory supervision includes monitoring the licensees’ progress on the action plans and adopting a subsequent standpoint as to whether the measures fulfil the requirements imposed by SSMFS 2008:17 to the extent that this is feasible. An overall assessment of compliance is however not possible to conduct at the present time, since both analysis work and modifications to the facilities remain to be done in the form of site modifications and rebuilding work. Some of the requirements imposed by SSMFS 2008:17 can govern several measures, and one measure can contribute to fulfilment of several requirements. Thus, this situation makes it more difficult to assess the overall fulfilment of requirements for each facility until all the measures have been taken and they have been reviewed by SSM.

SSM’s regulatory supervision also includes reviewing the licensees’ safety analysis reports containing the information presented on how the requirements imposed by SSMFS 2008:17 were interpreted and applied for not only parts of the facilities deemed as fulfilling the requirements in their present construction, but also parts that are to be modified and rebuilt. In several cases, reviews here have led to the Authority drawing the conclusion that the licensees’ safety analysis reports on how requirements are applied and verified are all too generally worded to give a clear and complete understanding of compliance with the requirements. Reviews to date have also shown that the licensees’ interpretation of several of the requirements does not agree with SSM’s point of view on the regulations’ implication. For example, the licensees have in several instances not taken the opportunity to diversify auxiliary systems, including electrical power systems and control systems, to a sufficient extent for fulfilment of the requirement on measures to prevent common cause failures as per Section 10 of SSMFS 2008:17. This mainly pertains to the newest reactors, whereas the oldest ones have, through the extensive modernisation work undertaken, achieved a greater degree of diversification. SSM will for this reason be imposing requirements on the licensees to improve their safety analysis reports' descriptions of their compliance as well as to perform physical measures to increase their level of diversification.

2.4.2. Impact on safety

The measures taken and planned ensuing from SSMFS 2008:17 strengthen protective functions of the barriers, mainly through increased redundancy and separation, which is the primary purpose of the measures. Also, when fully implemented, the measures imply a strengthening of the defence in depth of all the facilities. Another safety-related consequence of these measures, other than the purely physical modifications of the facilities, is the improved level of knowledge about the facilities' characteristics that the analyses vis-à-vis the legal requirements of SSMFS 2008:17 have implied among the licensees, as well as the fact that the technical documentation concerning the facilities has been improved. These factors are crucial prerequisites for ensuring safe nuclear power plants, as required by SSM through regulation SSMFS 2008:1.

Ringhals and Oskarshamn have performed quantitative evaluations by using probabilistic safety analyses (PSAs) of measures taken and planned.

However, most of the measures are of a kind having little impact on the total frequency of core damage[8]. This is partly because there are several important safety factors where the PSA evaluation has its limitations. One of these is an evaluation of facility robustness, that is, the capability to deal with unforeseen events. The same difficulty arises when evaluating the different levels of the defence in depth. Furthermore, not all measures are reflected by the PSA models, and the PSA methods used today also do not evaluate the long-term sequence of an accident in the form of the situation at hand after the first few days. Consequently, these factors contribute to a PSA evaluation not painting the full picture of the measures' impact on safety.

Despite these limitations, the quantification performed by Ringhals and Oskarshamn using PSA indicates that the frequency of core damage was reduced by more than half at Ringhals and dropped by up to two-thirds in Oskarshamn's reactors.

Forsmark is of the opinion that the contribution to reducing the frequency of core damage and discharges is negligible and has for this reason not performed a quantitative safety evaluation of the measures. In the assessment of SSM, based on the information presented by the other licensees, a significant reduction in the frequency of core damage has been achieved.

[8] The frequency of core damage refers to the anticipated frequency of damage to the core. All events bringing the reactor core temperature to an excess of 1204 degrees Celsius are included in the definition of core damage. Many of the event sequences that lead to core damage do not develop further into a core melt ('recovery sequences'), meaning that the frequency for core melt is lower than the frequency for core damage.

2.5. Need for further measures

2.5.1. Independent coolant makeup system

SKI's work to draw up regulations concerning the design and construction of nuclear power reactors involved considering whether to require measures making it possible to supply water to the reactor pressure vessel during a severe accident by connecting to a source of water located outside the reactor containment. Thus, the current situation does not offer any potential for supplying the reactor pressure vessel with water in Swedish facilities in the event of a total loss of electrical power. What was discussed at the time was activating injection independently of the reactor protection system and having a separate power supply. Against the background of insights from safety analyses and research and development projects, SKI assessed that this kind of measure would reduce the risk of core melt in connection with a serious accident by means of a coolant makeup system completely independent of the other emergency cooling systems. If core melt took place nevertheless, the risk of vessel melt-through would reduce considerably since it is very likely that water would reach this area and cool and thus contain a core melt in the reactor pressure vessel. This kind of independent coolant makeup system would imply a significant strengthening of the facilities' defence in depth by reducing the probability of core melt and vessel melt-through, at least when it comes to an accident scenario involving a total loss of all power systems without battery back-up.

Explicit requirements on independent coolant makeup systems with an external source of water were nevertheless not incorporated in SKIFS 2004:2, beyond the more general requirements for instance applying to diversification. At the time, there were also strong objections from parts of the nuclear power industry as they assessed that this kind of system in certain respects could have a negative influence on the safety of reactor containments in the form of a changed water balance. Therefore, SKI determined that further investigation was necessary. This kind of investigation was completed in March of 2009. Among other things, this investigation showed that:

1. An independent system for supplying water to the reactor vessel from a source of water outside the reactor containment significantly reduces the risk of core melt and consequently the risk of vessel melt-through during a loss of ordinary coolant makeup.

2. An independent coolant makeup system reduces the risk of serious core damage and may also prevent or delay vessel melt-through even if the system is started up when limited core damage has already taken place.

The investigation also helped SSM to draw the conclusion that there was no negative impact on safety. As a result, the Authority proceeded over the course of 2010 and 2011 to draft the requirements, but resolved following the accident at Fukushima Dai-ichi to await identification of additional lessons learned that could have an impact on the design of independent coolant makeup systems at Swedish nuclear power plants.

SSM has now resumed its preparations for imposing requirements on independent coolant makeup systems. SSM intends to also require the systems to be designed and set up together with protection that, combined with the other protective measures at the facility, allows the additional coolant makeup function to be maintained to the extent needed during a design basis antagonist threat over the period of time determined by the Authority.

2.6. A comparison with requirements and safety improvement measures in other countries

As described by section 2.1, the work on producing the regulations concerning the design and construction of nuclear power reactors originated from domestic and foreign operating experience, safety analyses, findings from research and development projects as well as the development of the IAEA's safety standards in the early 2000s. Since this time, additional lessons have been learned and national regulations and international standards have been updated. In addition, work is ongoing in certain countries to design and build modern Generation III reactors. This is why the framework of this investigation has also included conducting a comparison of the requirements imposed by SSMFS 2008:17 with the requirements ensuing from new national frameworks and international standards.

What's more, modernisation work and safety improvements at nuclear power plants have been examined in a number of countries. The aim of this review and the comparison between regulatory frameworks was to identify a possible need for additional safety improvements that also could be relevant to implement at nuclear power plants in Sweden in order to increase the safety margins against unforeseen events at ageing facilities. These could either be measures not ensuing from the requirements of SSMFS 2008:17 or, alternatively, measures not identified when applying the requirements of SSMFS 2008:17.

2.6.1. Comparison between Swedish and international regulatory frameworks

This comparison covers regulations and guides mainly directed at the design and construction of nuclear power reactors in Finland, the United Kingdom, Canada and the United States. Comparisons have also been made with the IAEA's new standard for the design of nuclear power plants and with the safety targets produced by WENRA.

This comparison does not specifically point out additional measures for the purpose of modernising Swedish nuclear power plants beyond what ensues from, or may be deemed as ensuing from, the requirements and the general advice contained in SSMFS 2008:17. On the other hand, this comparison indicates that the Swedish requirements are expressed far more briefly, the level of detail is significantly lower and that the requirements imposed are both less clear-cut and specific. For example, the following areas are regulated more specifically and in more detail by the countries studied in addition to the safety standards of the IAEA:

- Ageing and extensions of operating permits
- Siting of nuclear power plants
- Construction and commissioning of a nuclear power plant
- Man-Technology-Organisation Factors (including safety culture)
- Learning from operating experience

More clear-cut requirements are also contained in the safety standards of the countries studied and those of the IAEA in terms of:

- Safety analysis and risk analysis (probabilistic safety analysis)

- Verification of safety (e.g. applying ‘safety cases’)
- Fire protection
- Electrical systems
- Instrumentation and control equipment (including computer-based systems)

The comparison with the safety targets defined by WENRA for example shows that the interface between nuclear safety and security is not as clear in the Swedish requirements. It is urgent to regulate this area for several reasons. This need for instance arises in connection with modern software-based equipment being implemented at pre-existing facilities, and this in turn creates a need for a strategy and managing this equipment taking into account aspects of both safety and physical protection. A more detailed account of the requirements compared is presented in the report [2].

It was also established from the IRRS review performed of work activities in Sweden and at SSM in the fields of nuclear safety and radiation protection that the Authority needs to create a regulatory framework that is clearer and more comprehensive. The experiences from applying e.g. SSMFS 2008:17 have demonstrated the same need. See also information about the IRRS in section 4.4 and SSM’s planned revision of its regulations and guides in section 4.5.4.

The ongoing investigations at SSM as a basis for developing its regulations will consequently reveal which additional analyses and measures that Swedish nuclear power plants may need to carry out in order to (for example) increase their safety margins against unforeseen events at ageing facilities. What’s more, the stress tests performed owing to the accident at Fukushima Dai-ichi have indicated the need for additional analyses and measures. These are described in more detail in section 2.8 below.

2.6.2. Safety improvements made to nuclear power reactors in Europe

A review of safety improvements made at nuclear power plants in Finland, France, Switzerland and Germany has been conducted with the aim of identifying measures that could also be relevant at Swedish plants.

These safety improvements were initiated in different ways, for example originating from a regulatory authority’s requirements, such as conditions for a renewed operating permit, or made on a licensee’s initiative without an express requirement imposed by an authority.

When making a comparison, it appears that Sweden is in the forefront as far as concerns the scope and to some extent its progress in the area of safety improvements. Sweden has for instance been an early adopter when implementing extensive modifications such as the accident mitigation functions.

Most of the identified safety improvements studied as part of this review were ones that the various countries and respective types of reactors had in common. As the original design varies from one reactor to another, however, there are considerable differences in terms of both the scope and number of identified safety improvements. It has been established that physical separation, redundancy and diversification (whose aim is independence) are aspects that have been generally strengthened at the oldest reactors, whereas the conditions for these areas are better at the newer plants, so these plants have not required as extensive safety improvements.

Because the information available on the countries' implemented safety improvements has been limited and expressed very generally, it has been difficult to identify non-performed safety improvements at Swedish nuclear power reactors. It nevertheless appears as though other countries have generally devoted more work than Sweden on the capability to deal with natural phenomena and other events that might take place inside and outside a facility (earthquakes, flooding, fire, disruptions in the power supply system, including a total loss of electrical power, also called 'station blackout'). Furthermore, the review identified relevant safety improvements applicable to independent cooling of the reactor core in a long-term sequence, ageing, dealing with hydrogen, the capability to supply the seals of the main circulating pumps with water during a common mode failure in the primary pumps' power supply, in addition to the fuel pools' cooling and integrity.

However, in connection with the stress tests of Swedish nuclear power plants, many of these aspects were also considered on the part of these plants and will be taken into account in the form of the action plans to be decided upon. See also section 2.8 below.

2.7. General information about measures taken as a consequence of the accident at Fukushima Dai-ichi

The accident at the Fukushima Dai-ichi nuclear power plant in Japan as a consequence of a severe earthquake on 11 March 2011 and subsequent tsunami has led to varying measures taken in many countries around the world and by international organisations. The International Atomic Energy Agency (IAEA) has for example convened meetings of experts in different fields, an extraordinary meeting in August 2012 within the framework of the Convention on Nuclear Safety, and has launched projects for reviewing its safety standards. Special working groups have been set up within the framework of the OECD Nuclear Energy Agency's (NEA) government and research co-operation. Other European and international bodies have also convened meetings and set up working groups.

Governments and parliaments around the world have tasked their respective national radiation safety authorities with investigations, identifying lessons learned and taking measures against the background of the accident.

The following is a brief account of progress in Sweden and elsewhere in Europe:

On 22 March 2011, the Swedish Radiation Safety Authority (SSM) pointed out in a written communication to the licensees the importance of immediately launching work to identify lessons learned from the situation with the aim of assessing any further radiation safety measures that might be necessary at Swedish nuclear power plants and at Clab, the facility for interim storage of spent nuclear fuel.

Following an extraordinary meeting, on 24 and 25 March 2011 the Council of the European Union declared that Member States were prepared to begin reviewing safety at nuclear facilities by means of a comprehensive assessment of risk and safety ('stress tests') on the part of reactors in the European Union. WENRA then drew up (e.g. with the assistance of experts from SSM) and proposed specifications for these tests' scope and orientation. ENSREG subsequently agreed on specifications for the stress tests. These included having the licensees perform the tests, which were then to be reviewed, first by the national radiation safety authority and subsequently by experts in European peer review teams. These specifications were approved on 25 May 2012.

On 12 May 2011, the Swedish Government resolved to supplement the previous assignment to SSM to also encompass: a.) submitting a comprehensive report on the stress tests of the relevant Swedish nuclear facilities to be performed on the basis of ENSREG's specifications, and b.) presenting other lessons learned from the accident at Fukushima Dai-ichi.

On 25 May 2011, SSM resolved to enjoin Swedish nuclear power plants and Clab, the central interim storage facility for spent nuclear fuel, to conduct comprehensive risk and safety assessments of their resilience against earthquakes, flooding and extreme weather conditions, in addition to prolonged power failures and loss of heat sink. This decision also implied the licensees analysing and evaluating the facilities' emergency preparedness and pre-planned measures for managing severe accident sequences assuming extensive destruction of surrounding infrastructure as well as radioactive contamination at the site.

On 31 October 2011, the licensees reported on the results of their stress tests, and on 15 December 2011, SSM's review was completed and the Authority presented an interim report to the Government concerning the stress tests.

On 29 December 2011, the Swedish national report on the stress tests was submitted to ENSREG and other bodies.

In the spring of 2012, a European peer review of the results (i.e. the national reports) from all countries was performed. These countries had undertaken to perform stress tests in accordance with ENSREG's specifications.

The review reports from the European peer review were approved on 26 April 2012. In connection with publication of these reports, a statement by ENREG was also issued on the results of the stress tests and a planned follow-up of these results.

On 26 April 2012, SSM resolved to enjoin the licensees to, no later than 15 September of the same year, report on their plans to use the stress tests performed and reviews of these as a platform for further strengthening the facilities' resilience against earthquakes, flooding and extreme weather conditions, in addition to prolonged power failures and loss of heat sink. This decision also encompassed the licensees' plans to further improve their emergency preparedness and pre-planned measures for managing severe accident sequences.

On 16 July 2012, SSM resolved to enjoin the Swedish Nuclear Fuel and Waste Management Company (SKB) to submit corresponding plans on the part of Clab by 1 November.

During a meeting held on 28 and 29 June 2012, the Council of the European Union urged Member States to take the measures necessary to ensure 'complete and quick implementation' of ENSREG's recommendations following the stress tests.

On 1 August 2012, ENSREG published a plan for following up the stress tests: "Action plan – Follow-up of the peer review of the stress tests performed on European nuclear power plants". This plan states that all EU Member States must produce a national action plan in 2012 for dealing with the lessons learned from the accident at Fukushima Dai-ichi, the deficiencies and recommendations identified as part of the peer review of the stress tests, as well as the areas identified during the extraordinary review meeting held under the Convention on Nuclear Safety in August 2012. National action plans are to have been produced and published by the end of 2012. These plans are then to be discussed and reviewed together during a meeting (a workshop) held by ENSREG in March 2013.

On 15 September 2012, the licensees submitted their action plans in the light of the stress tests performed and SSM’s injunction.

On 3 October 2012, ENSREG published a compilation of the peer review of the stress tests performed in Europe: “Compilation of stress test peer review recommendations and suggestions”.

On 4 October 2012, the European Commission submitted its integrated report from the stress tests and measures taken and planned to the Council of the European Union and the European Parliament: “Communication from the Commission to the Council and the European Parliament” (COM(2012) 571 final).

2.8. Stress tests performed

2.8.1. The aim of the tests and licensees’ comprehensive risk and safety assessments

The aim of the comprehensive risk and safety assessments (‘stress tests’) resolved by SSM was to, in the light of the accident at Fukushima Dai-ichi, a.) study whether present safety analyses on the part of Swedish nuclear power plants and Clab remain valid or whether additional information is needed for fulfilment of the requirements on facility design, and b.) evaluate more serious conditions than those taken into account during the design and construction of the facilities. These studies examined and evaluated safety margins and cliff-edge effects for the purpose of assessing the robustness of the nuclear power plants and identifying a possible need for safety improvements and/or further analyses.

The findings from the licensees’ comprehensive risk and safety assessments of nuclear power plants show that the Swedish facilities are robust, but that there is a need to further improve the level of safety. Many of the areas of improvement identified imply that analyses conducted earlier need to be supplemented, or that new analyses need to be conducted before adopting a standpoint as to whether a measure must be taken, and in this case, in what way.

Apart from the need to conduct additional analyses, the need for more concrete measures at the nuclear power plants has also been identified, for example installation of equipment or improved emergency response management by allocating more resources and/or approving new procedures. These kinds of measures also presuppose further analyses as a platform for design and implementation.

Safety functions at the central interim storage facility for spent nuclear fuel, or Clab, are passive ones, implying that they need no electrical power to perform their functions. The safety functions consist of fuel cassettes and cassette racks, storage pools for fuel and the specific rock cavern/storage building itself. Clab also has a passive safety system that makes it possible to inject water from various sources to the pools in the storage area.

The results from the licensee’s comprehensive risk and safety assessments of Clab indicate that this facility is robust and capable of withstanding the events for which the facility has been designed, also that the safety margins are sufficient for many situations. Conceivable accident sequences at Clab are relatively slow, giving the personnel time for consideration to allow for countermeasures. Should several events take place at the same time as the facility is in an unfavourable operational mode and countermeasures fail, then small discharges of radioactivity cannot be ruled out.

2.8.2. SSM's view on comprehensive risk and safety assessments of nuclear power plants

In the view of SSM, the licensees' comprehensive risk and safety assessments have culminated in the identification of a number of needed safety improvements which can further improve safety at Swedish nuclear power plants. SSM has nevertheless identified several safety improvements that are needed in addition to those identified by the licensees. Also, SSM has identified omissions in the comprehensive risk and safety assessments that may be viewed as deviations from ENSREG's specifications.

The comprehensive risk and safety assessments have also identified deficiencies and deviations in relation to the requirements applying to safety analysis. In these cases, SSM intends to order the licensees to take action so that the facilities fulfil the requirements. SSM nevertheless assesses that none of the deficiencies nor deviations currently identified are of such a nature that the continued operation of the facilities is put into question.

In the assessment of SSM, the measures to increase the level of safety that are gradually being implemented at Swedish nuclear power plants, both earlier and in later years as a consequence of the requirements imposed by SSMFS 2008:17, have contributed to these plants now being considered to be robust. The comprehensive risk and safety assessments have for instance demonstrated the importance of the consequence-mitigating systems, primarily the accident filters, for dealing with unforeseen events. In the assessment of SSM, if a situation similar to that at Fukushima Dai-ichi should arise at a Swedish nuclear power plant, the accident systems should mitigate the event sequence and minimise any potential discharge to the environment. This is because the accident filters have a purification function that captures a large proportion of the radioactive substances that otherwise could be released into the environment in connection with a severe accident at the same time as they discharge heat from the reactor into the atmosphere.

Earthquakes

SSM assesses that the licensees have not taken the measures required under the Authority's regulations for certain reactors. It has for example not been fully demonstrated that important functions needed to bring the reactors Oskarshamn 2, Forsmark 1, Forsmark 2, Ringhals 2, Ringhals 3 and Ringhals 4 to a safe state will perform as intended during and after an earthquake.

Also, the licensees need to complete the in-depth analyses required for evaluating the safety margin for safe shutdown and implementation of the improvements identified in the updated comprehensive risk and safety assessments. As far as concerns the reactors at Forsmark and Ringhals, a more detailed analysis also needs to be conducted in terms of earthquake-induced flooding.

Flooding

All the nuclear power plants are capable of withstanding a rise in sea water level of 3 metres, which the licensees estimate has a probability of once per 100,000 years (10^{-5}/year). In the assessment of SSM, this estimate should be evaluated further.

Combination effects of waves and high water levels have not been taken into account for all facilities. This is why further analyses are needed to take these combinations into consideration as well as to shed light on potential dynamic effects in connection with flooding phenomena.

Extreme weather conditions

The comprehensive risk and safety assessments demonstrate the nuclear power plants' resilience against the conditions that might arise at the plants as a result of different kinds of extreme weather conditions. The comprehensive risk and safety assessments nevertheless show that a number of areas contain major uncertainties or for some other reason should be investigated further to make it possible to identify opportunities to further strengthen the facilities' protection in connection with these events. For example, the procedures for the working staff in terms of requisite measures in the event of large quantities of precipitation and extreme temperatures should be reviewed. Also, no in-depth analyses of combinations of different weather phenomena have been conducted, such as extreme snowfall together with extreme winds.

Furthermore, it has been established that there is a lack of detailed and thorough descriptions of how the nuclear power plants are impacted in connection with possible ice storms. One engineering assessment, however, is that an extreme ice storm might cut offsite power and risk blocking ventilation systems and hampering access to the site. The fact that in-depth analyses have not been conducted is assessed as a deficiency in relation to current regulations and must consequently be performed.

Consequence-mitigating systems

The comprehensive risk and safety assessments demonstrate the importance of consequence-mitigating systems, for example accident filters and the independent functions for containment spray. This mainly applies in connection with power failures and a loss of heat sink, or a combination of both these events. However, it has been observed that there are uncertainties in the analyses of consequence-mitigating system performance in a long-term sequence, so it needs to be ensured that these systems are capable of performing during long-term accident sequences in addition to all the conditions applying to the scenarios in which the systems are credited. This for example applies to the conditions arising if these systems are used for transferring heat from the reactor core to the atmosphere.

Power failures

All Swedish nuclear power plants have alternative reserve power systems in the form of gas turbines within or close to the site. However, these reserve power systems have not been safety classified, meaning that lower requirements on quality and testing apply than for safety systems. As the comprehensive risk and safety assessments indicate that an alternative reserve power system could be crucial during a sequence of events where all offsite power and ordinary reserve power is unavailable, the need to strengthen these systems should be investigated, particularly when considering situations where several reactors are affected simultaneously.

In the event of a loss of all alternating current (i.e. a loss of offsite power in addition to loss of ordinary and alternative reserve power), only the power systems with battery back-up for instrumentation and manoeuvring of components remain operational. At the present time, requirements are imposed on the batteries working for one to four hours, although analyses and support documentation show that they can work for a longer period of time. Thus it has been deemed crucial to review the potential for increasing the current battery capacity by qualifying the batteries for longer periods of operation or, alternatively, by disconnecting the batteries from non safety-critical equipment while also examining the potential to recharge the batteries using mobile equipment.

It must be possible to use mobile equipment if there is a loss of all alternating current, but the capacity and number of mobile units are insufficient for all kinds of events, particularly if several reactors are affected simultaneously. This is why it is considered essential that the licensees take stock of their mobile units to ensure that they have an adequate quantity of units, that they offer sufficient capacity and are available in the event of severe accidents.

The comprehensive risk and safety assessments also show that there may be a need to refill lubricant within a few days at some facilities, which is why a sufficient supply of lubricant should be ensured at the site.

Loss of main heat sink (failed removal of heat to the sea or into the atmosphere)
All Swedish nuclear power plants are dimensioned to be brought to a safe state if the seawater inlet is blocked, also to keep the plant in this state. It was nonetheless shown from the comprehensive risk and safety assessments that this has not been fully verified as far as concerns Ringhals 3 and 4, so this work remains to be done.

Simultaneous blockage of both inlet and outlet channels has not been taken into account previously as part of the analyses of these plants and the comprehensive risk and safety assessments now show that these conditions necessitate a number of manual measures. It has been established that an in-depth analysis of manual measures that may be necessitated by accident sequences that have been taken into account needs to be performed, in addition to an evaluation of available resources. These analyses should consider access to the facility on the basis of assumed accident sequences and their potential impact on the work environment.

The comprehensive risk and safety assessments now also demonstrate the major significance of independent core cooling functions, where both permanent and alternative systems as well as mobile units raise the level of the facilities' safety and robustness. For the purpose of ensuring the availability and performance of these systems, in-depth analyses should be performed to evaluate present independent core cooling functions and to identify a potential need for additional improvements or implementation of new systems. See also section 2.5.1.

In order to maintain cooling of the fuel pools in accident situations, manual measures are needed; at the same time, however, lessons learned from Fukushima Dai-ichi show that access to reactor buildings can be hampered during severe accidents. This is why it is considered essential that the licensees evaluate the potential to implement alternative solutions for cooling of fuel pools by implementing both permanent installations and mobile units. A key prerequisite in connection with these investigations is to take into consideration the personnel's capability to carry out potential manual measures in connection with these events/accidents.

Emergency response management and emergency preparedness
The comprehensive risk and safety assessments demonstrate the importance of the consequence-mitigating systems, where the accident filters are key. In an accident situation where residual heat removal has failed and the reactor core is melting through the reactor vessel, the pressure in the containment will rise until valves to the accident filter open and relieve the pressure from the containment into the atmosphere. This filter has been designed so that a considerable proportion of the radioactive substances that may be

present in the gases passing through the accident filters are captured, thus largely preventing ground contamination.

The accident filters were originally designed for 24 hours of operation without operator actions. As the lessons learned from the accident at Fukushima Dai-ichi have demonstrated that accident sequences can be prolonged and that it can be difficult in these situations to carry out manual actions within 24 hours, the licensees need to evaluate the accident filters in terms of long-term operation.

The requirements imposed on the accident systems’ batteries, which are used for maintaining surveillance and manoeuvring, stipulate that they must be available during initial accident sequences and subsequent recharging must be possible in an easily accessible way. However, the facilities’ design took neither account of accidents affecting several reactors at the same time nor serious destruction at the site as a result of natural phenomena. In order to ensure capability to monitor and manoeuvre accident systems, present battery capacity and related recharging possibilities need to be reviewed. However, it should be noted that the function of accident systems is passive and is consequently considered to be operational without available batteries.

In Sweden, work has long been underway to develop the facilities for the purpose of preventing hydrogen explosions. It has nonetheless been established that the licensees have not conducted a detailed and thorough study of the risk of hydrogen leakage to the reactor building, which in fact did occur from the reactors of Fukushima Dai-ichi. For this reason, the licensees must investigate these risks further. Above all, these investigations should focus on the risk of hydrogen accumulation in reactor buildings, as well as the need for additional monitoring to assist operators and other working staff. Beyond this, dealing with hydrogen over a long-term perspective needs to be taken into account.

Strategies for emergency response management are at the present time oriented at sequences where the consequence-mitigating systems protect containment integrity and thus prevent large and uncontrolled radiological discharges into the environment. Lessons learned from the accident at Fukushima Dai-ichi nevertheless indicate that pre-planned strategies are also needed covering accidents involving failure of the containment function and where considerable releases of radioactive materials are unavoidable.

When updating existing strategies for emergency response management, an in-depth analysis of the accident response organisation’s structure and staffing also needs to be performed to ensure that it is capable of dealing with all situations, in particular situations where several reactors are affected simultaneously.

The above-mentioned results are described in more detail in SSM’s report [2] and memorandum [9].

2.8.3. SSM’s view on comprehensive risk and safety assessments of Clab

A number of safety improvements have been identified to strengthen the Clab facility’s resilience against extreme situations and its capacity to deal with accidents. Examples of these include a need for in-depth analyses and tangible measures. Here, safety improvements can comprise fitting new equipment, improving procedures and instructions for severe accident situations, analysing the resilience of the facility’s conventional buildings, evaluating the capacity of various cooling systems as well as a more thorough survey of support documentation.

Earthquakes
The licensee's description shows that the components and structures of Clab designed to withstand an earthquake have been verified on the part of the earthquake taken into account during the facility's design and construction. When it comes to the evaluation of safety margins, the results show that the structures' resilience is limited against earthquakes beyond the levels taken into account during the facility's design and construction.

Loss of electrical power
In the event of a loss of offsite power, a reserve power generator ensures that handling of fuel can be completed and that the facility can be brought to a safe state. Clab only has one reserve power generator, however, so the need for additional reserve power must be evaluated further.

Loss of main heat sink (cooling)
If there is a loss of cooling, there is usually an extended period of time before the fuel risks becoming uncovered. There is also a possibility of pumping water into the pools via a passive safety system.

The licensee has however not provided a full account of how the movement joints between the storage areas are affected over a long-term sequence in excessive operating temperature, nor of the safety margins in effect during these conditions, for which reason further analyses need to be performed. Similarly, the licensee needs to review operating limits in connection with isolation of pools in the hall for incoming shipments.

Since the commissioning of Clab, one crack in the concrete of a spent fuel pool has been found. The licensee's report on its stress tests states that the pool, despite this situation, fulfils the applicable requirements and that the reinforcement is unaffected. In the assessment of SSM, however, further analysis of the crack's impact is necessary to support this conclusion.

Emergency response management and emergency preparedness
In most cases, Clab's current pre-planned measures for emergency response management presuppose access to water, or at the very least, reserve power. Improving the equipment for ensuring access to these functions would imply a considerable improvement to the capability of the accident response organisation to bring the facility to a safe state.

It has not been possible to verify the resources needed in the event of extreme situations that simultaneously affect nearby nuclear power plants. This consequently needs further investigation, particularly as regards access to the resources of rescue services.

2.8.4. Results of the peer reviews of the stress tests of European NPPs

The results from the European peer review of the comprehensive risk and safety assessments, organised by ENSREG, have been published both in the form of a summary review report for all the countries that undertook to perform the European stress tests, as well as in the form of review reports directed specifically at each country. These reviews are used as a platform to give recommendations to regulatory authorities and their collaborating organisations, and joint recommendations are provided to assist with the national action plans to be produced by the end of 2012. Also, the peer review has submitted recommendations specifically for the countries.

The European peer review:

- Recommends WENRA to set up a team of experts to develop guidance on how to conduct an analysis on the impact from natural phenomena and how to evaluate safety margins beyond the original design assumptions and cliff-edge effects
- Has drawn the conclusion that periodic safety reviews of facilities (PSRs) have proven to be an important tool for maintaining and improving facility safety and robustness. This is why ENSREG is recommended to focus on the importance of the periodic safety reviews of the facilities. In particular, it is emphasised that natural phenomena and critical functions are to be evaluated as often as deemed appropriate, though at least every tenth year
- Recommends that supervisory authorities ensure that all necessary improvement measures that have been identified in terms of the containment function are implemented as soon as possible. Examples of these safety improvements include pressure relief of the reactor system in order to prevent melt-through of the reactor vessel under high pressure, preventive measures against hydrogen explosions in addition to preventive measures against subjecting the containment to extremely high levels of pressure
- Recommends that supervisory authorities ensure that safety improvements are implemented to prevent accidents and mitigate potential consequences from extreme natural phenomena. Examples of these safety improvements include bunkered systems, instrumentation and communication equipment, mobile equipment protected against extreme natural phenomena, emergency response centres protected against extreme natural phenomena and contamination, and also rescue services with the necessary equipment quickly arriving at the site and providing assistance in connection with long-term accident sequences

The European peer review also recommends that the national supervisory authorities take into consideration a number of safety aspects when compiling the national action plans to be produced by the end of 2012. There are around 40 safety aspects that can be summarised as follows:

- Appropriate methods of analysis
- Natural phenomena to be taken into account
- Simultaneous events in different reactors at the same site
- Environmental qualification of equipment
- Installation of new instrumentation and warning systems
- Internal and external communication systems
- Proposed system solutions on a principal level
- Need for procedures and exercises
- The personnel's work environment in connection with an accident
- Radiation protection of workers
- Rescue services ready for action
- Feasible measures on the part of fuel pools
- Using mobile equipment
- Dealing with hydrogen

The European peer review also directed some recommendations specifically at Sweden in the form of several measures that should particularly be taken into account. These include:

- Evaluating the methods of analysis applied and input data for earthquakes
- Considering implementation of probabilistic safety analyses of external flooding
- Conducting in-depth analyses of extreme weather conditions in accordance with internationally accepted methods
- Reinforcing gas turbines
- Improving the capability to fill diesel oil in diesel tanks during extreme conditions

- Reviewing the capability to use lubricant from one reactor for equipment inside another reactor
- Increasing battery capacity
- Fitting pipes so that fire water can be supplied to the fuel pools
- Reviewing procedures for the capability to cool the reactor core using alternative sources of water, also ventilating hydrogen
- Taking into account long-term scenarios, particularly in terms of the accident filters' function

2.8.5. Measures taken and planned against the background of the Fukushima Dai-ichi accident

Since the accident at Fukushima Dai-ichi, a number of measures to increase the level of safety have been taken at Swedish nuclear power plants. These measures were mainly identified in connection with the comprehensive risk and safety assessments (stress tests) of Swedish nuclear facilities and in connection with investigation work linked to the licensees' international forum, the World Association of Nuclear Operators (WANO).

SSM's investigative report [2] and memorandum [9] summarise the approximately 60 measures that have been taken or are planned for completion by 31 October 2012, in addition to the licensees' assessments as to how the respective measure has had an impact on safety. These measures are of a relatively straightforward nature, such as updating procedures, general checks of organisation and cleanliness as well as preparatory investigations for later implementation of more extensive measures and plant modifications.

The investigative report [2] and memorandum [9] also summarise the plans of Swedish nuclear power plants, which were reported to SSM on 15 September in accordance with the Authority's injunction, for these more extensive measures and plant modifications. These plans are based on deficiencies and areas of improvement identified from the stress tests and subsequent reviews. The plans will serve as the basis of the national action plan to be produced by SSM by the end of 2012 and presented to ENSREG. The plans of the nuclear power plants, the results from SSM's reviews of the stress tests and the recommendations from the European peer reviews will also serve as the basis of the facility-specific injunctions concerning measures that the Authority intends to take a decision on in early 2013.

SSM also intends to take a decision on measures at Clab using the corresponding input after the licensee has presented its action plan, which is to be submitted to the Authority by 1 November 2012.

2.9. Need for further investigation and research owing to the accident at Fukushima Dai-ichi

As described in previous sections, the accident at Fukushima Dai-ichi led not only to immediate action, but also to updated comprehensive risk and safety assessments (stress tests) of nuclear power plants in many countries. The accident has also raised questions necessitating more in-depth investigations and research. These areas have been discussed nationally and through international fora such as the IAEA, CNRA and CSNI within OECD/NEA and WENRA. The areas identified for further investigation and research include:

- Methods for evaluating natural disasters such as design basis events and dealing with cliff-edge effects
- Evaluation of secondary effects from earthquakes, such as flooding and fires
- The availability and robustness of electrical systems
- Accident sequences and the applied codes' capability to compute and predict sequences
- Improved reliability of instrumentation during accident sequences
- Operators' actions in situations of extreme stress during accident sequences
- The availability of control rooms and emergency control rooms during long-term accident sequences involving several production units at the facility site
- Decision-making processes and communication between the relevant stakeholders (licensees, rescue services, authorities) during severe and long-term accident sequences

SSM already has ongoing and relatively extensive investigative and research partnerships within the international fora. The Authority intends to intensify these efforts and will take part in many of the investigations and research projects launched against the background of the accident at Fukushima Dai-ichi.

For more information about investigations and research as a consequence of this accident, please see SSM's investigative report [2] and memorandum [10].

Other questions raised relate to the authorities' access to reliable information and facility data in accident situations for performing their tasks as expert bodies and providing advice to rescue services and other public bodies whose task is to safeguard life, health and the local environment. A key component of this work is enabling instant access to facility data at the authorities' emergency response centres. Several radiation safety authorities internationally have this capability, and in Sweden, preparations are now being made so that technologies and organisations will be capable of this kind of instant communication in actual or potential accident situations. This will be achieved by means of voluntary agreements between the licensees and SSM pending legal references as proposed by the Inquiry into coordinated legislation in the fields of nuclear engineering and radiation protection [11].

2.10. Brief information about measures for strengthening the facilities' physical protection

In parallel with the measures taken and planned for safety upgrades to the facilities, projects have been ongoing for several years now that focus on improving their protection against malicious acts. These improvements are being made owing to the Authority's regulations, SSMFS 2008:12, concerning security, which entered into force on 1 February 2009 (formerly issued as SKIFS 2005:1). Since then, extensive measures have been taken at all nuclear facilities, nuclear power plants in particular, with the purpose of improving this protection. The results are strengthened protection in terms of detection and verification of unauthorised access, access protection in the form of strengthened barriers, access control and security checks of individuals, goods and vehicles in addition to strengthened protection of certain spaces at the facilities.

A review of the capability of licensees and the State to provide protection against antagonistic threats, however, indicates that protection against sabotage needs to be strengthened further. For this reason, investigations are in progress at SSM to ascertain which additional measures are needed; also, amendments to the Authority's regulations in the area are under preparation.

Some of these measures were also identified from the stress tests of the nuclear power plants. This means that it is also important to take the requirements imposed on security into account in connection with design and construction work with the purpose of improving safety because of deficiencies identified from the stress tests. For instance, mobile standby systems need to be protected from impact from both antagonists and external events such as storms, high water levels and precipitation. At the same time, however, this protection against antagonistic threats must not hamper quick connection of the equipment when this is necessary.

2.11. Conclusions

The safety upgrades begun by the licensees when the Authority's regulations (at the time, SKIFS 2004:2, presently SSMFS 2008:17) concerning the design and construction of nuclear power reactors entered into force on 1 January 2005 are now showing progress so that the facilities will fulfil the requirements. However, the work on these measures proved to be much more complicated and time consuming than was foreseen when the licensees produced their proposals and the action plans were decided by the Authority. The licensees have also mentioned that their work taking the various measures identified a need for certain additional and to some extent other measures for fulfilling the requirements. Furthermore, some of the licensees have stated that running the modernisation programmes has been so wide-ranging that it has not been possible to take all the measures by the specified period of time while also maintaining high standards on quality.

These problems have led to the licensees applying (and receiving approval) for an extended deadline for when some of the measures are to have been taken, and consequently when their compliance with the regulation is to have been achieved. The longest postponement of the deadlines for measures is until 2015.

Up until 30 June 2012, altogether for the ten reactors' modernisation programmes, approximately 60 per cent of the resolved measures had been implemented. There are major differences between the reactors' progress, where for a few of the facilities, a great deal of work remains to be done. It is nevertheless important to point out that the measures are of varying safety importance and scope, which is why only comparing the number of measures taken and remaining does not provide an accurate picture of the overall safety improvement nor progress of the respective action plan.

SSM's regulatory supervision includes monitoring the licensees' progress on the action plans and adopting a subsequent standpoint as to whether the measures fulfil the requirements imposed by SSMFS 2008:17 to the extent that this is feasible. An overall assessment of compliance is however not possible to conduct at the present time, since both analysis work and modifications to the facilities in the form of site modifications and rebuilding work remain to be done. Some of the requirements imposed by SSMFS 2008:17 can govern several measures, and one measure can contribute to fulfilment of several requirements. Thus, this situation makes it more difficult to assess the overall fulfilment of requirements for each facility until all the measures have been taken and they have been reviewed by SSM.

SSM’s regulatory supervision also includes reviewing the licensees’ safety analysis reports, with the details presented on how the requirements imposed by SSMFS 2008:17 were interpreted and applied for not only parts of the facilities deemed as fulfilling the requirements in their present construction, but also parts that are to be modified and rebuilt. In several cases, reviews here have led to the Authority drawing the conclusion that the licensees’ safety analysis reports on how requirements are applied and verified are all too generally worded to give a clear and complete understanding of compliance with the regulations. Reviews to date have also shown that the licensees’ interpretation of several of the requirements does not agree with SSM’s point of view on the regulations’ implication.

SSM assesses that the measures taken and planned to fulfil the requirements imposed by SSMFS 2008:17 strengthen protective functions of the nuclear power reactors’ barriers, mainly through increased redundancy and separation, which is the primary purpose of the measures. Also, when fully implemented, the measures imply a strengthening of the defence in depth system on the part of all facilities. Another safety-related consequence of these measures, other than the purely physical modifications of the facilities, is the improved level of knowledge about the facilities’ characteristics that the analyses vis-à-vis the legal requirements of SSMFS 2008:17 have implied among the licensees. The technical documentation concerning the facilities has also been improved. These factors are crucial prerequisites for ensuring safe nuclear power plants.

SSM has, however, in the presently completed analysis of safety improvements drawn the conclusion that further measures will be needed in addition to those of the licensees’ action plans for fulfilment of the requirements of SSMFS 2008:17. The results of the comprehensive risk and safety assessments (stress tests) also indicate the need for measures in order to strengthen resilience against extreme natural phenomena, a loss of power and a loss of main heat sink. Furthermore, the facilities’ emergency preparedness and capability for emergency response management need to be strengthened in various respects. SSM will enjoin the licensees to carry out necessary safety improvements.

SSM has made the assessment that the nuclear power plants also need to be provided with systems for independent coolant makeup. This kind of system reduces the risk of core melt and thus also the risk of a melt-through of the reactor pressure vessel in the event of a loss of the ordinary coolant makeup. SSM is now drafting requirements on independent coolant makeup systems, and also intends to impose requirements on these systems’ design and setup having protection that, combined with the other protective measures at the facility, allows the additional coolant makeup function to be maintained to the extent needed during a design basis antagonist threat during the period of time determined by the Authority.

A review of the capability of licensees and the State to protect the facilities against antagonistic threats also indicates that protection against sabotage needs to be strengthened further. Investigations are in progress at SSM to ascertain which additional measures are needed and amendments to the Authority’s regulations in the area are under preparation.

Altogether, this signifies that the Swedish nuclear power industry is facing major challenges, including both continued analyses and continued measures for fulfilment of present and additional requirements imposed on safety and security. Consequently, SSM also needs to transition into imposing more clear-cut requirements while intensifying its regulatory supervision to ensure that the licensees show progress in their ongoing modernisation and safety improvement work, in addition to the measures being implemented in a way that is safe and that maintains an appropriate standard.

3. Analysis of ageing issues in connection with long-term operation of reactor facilities

3.1. Starting points

As with many facilities around the world, when the Swedish nuclear power plants were designed and constructed, a period of operation of approximately 40 years was assumed. This included performing design analyses and fatigue calculations with assumptions concerning a certain number of startups and shutdowns of the facility, other operational modifications, scrams and various kinds of transients during this period of time. Consequently, long-term operation (LTO) refers to operation beyond the period of time for which the facilities were originally designed and analysed.

This kind of long-term operation of nuclear power plants is being planned or has begun in many countries to meet their demand for electrical power and to meet the climate targets for emissions of greenhouse gases. In some countries, long-term operation is viewed as a way to fill the gap until today's reactors have been replaced by new ones or the emergence of some other new kind of production capacity. Swedish licensees have announced that they intend to operate the nuclear power plants for 50 to 60 years. Swedish nuclear power plants are presently 27 to 40 years old, counting from the point in time when they began routine operation. Oskarshamn 1, which began routine operation in 1972, is now approaching long-term operation, and in 2013, SSM will take a decision on its long-term operation.

Some countries, such as the United States, limit the licences to 40 years of operation. An extension requires new authorisation in the form of licence renewal. The American supervisory authority, the U.S. Nuclear Regulatory Commission (NRC), has up until June 2012 taken a decision on more than 70 licences for long-term operation of nuclear power plants. Other countries, such as Sweden, do not limit licences to a specific period of time. Instead, the licences are in effect as long as all requirements are fulfilled.

Long-term operation presents both licensees and regulatory authorities with new challenges, starting from the requirements imposed on safe operation during the extended period of operation. The organisation, resources and expertise of both licensees and supervisory authorities must be adapted to manage new safety issues that might arise in connection with long-term operation. Authorities' regulatory frameworks need to place a clearer focus on aspects of ageing, the licensees' ageing management activities as well as the need for gradual upgrades to increase the safety margins against unforeseen events at ageing facilities. Regulatory supervision also needs to place a greater focus on ageing management issues.

3.2. General information about ageing and ageing management

The topic of ageing nuclear power plants usually refers to the ageing of systems, components and building structures included in the barriers and the plants' defence in depth. This kind of ageing is a process in which the physical properties change in some respect over time or during use. This is why maintaining control over physical ageing presupposes good planning by the licensees through preventive measures, for example by replacing parts sensitive to damage, also close monitoring and regular inspections of the facilities' barriers and defence in depth systems, including subsequent corrective repairs when damage or other deterioration has been discovered.

However, other aspects of ageing also need to be taken into account, such as technological ageing of instrumentation and control equipment. Even if obsolete equipment is still working, it can be difficult to maintain and repair, or to find spare parts for since the equipment is no longer being manufactured or is no longer available on the market. Another topic of discussion is obsolescence of organisations and regulatory frameworks. However, these areas are not dealt with further in this report, other than in connection with periodic safety reviews. See also sections 3.5 and 4.6.

SSM imposes requirements on the licensees' ageing management activities in terms of physical and technological ageing and stipulates that an ageing management programme must be in place for this area. An ageing management programme can be viewed as a comprehensive coordination programme consisting of other maintenance and inspection programmes, such as a surveillance programme for reactor pressure vessels, and programmes for environmental qualification, water chemistry follow-ups and monitoring. Requirements are also imposed on the ageing management activities being subject to the licensees' management systems. The aim is to ensure long-term management of ageing issues and to as far as reasonably possible prevent degradation and other deficiencies from arising so that barriers, structures and safety systems no longer work as intended. The requirements imposed on ageing management and ageing management programmes apply generally, but obviously become of greater importance as the facilities age.

In its report [3], SSM has described ageing-related aspects of the analysis assumptions for operating Swedish reactors over extended periods of time, made an assessment on the factors related to ageing that will be decisive as to whether a reactor can be operated further over extended periods with a sustained level of safety, as well as proposed improvements to the Authority's regulatory supervision and regulations with long-term operation in mind. This is summarised in sections 3.3 to 3.6 below.

3.3. The general knowledge situation in terms of ageing and degradation mechanisms

The extensive research conducted nationally and internationally over the past 30 years or so has provided both licensees and supervisory authorities with sound knowledge about the degradation mechanisms that can give rise to damage inside nuclear power plants. This knowledge has brought about measures to prevent damage as well as the application of effective programmes for ageing management and inspections. An overall evaluation, which covers all cases of damage in mechanical components ever since the first facility was commissioned in Sweden, confirms that the measures taken to prevent and repair damage have had the impact intended. There is currently no clear trend showing an increase in cases of damage as the facilities age. The overall evaluation also shows that most of the damage occurring to date was discovered in time through the in-service inspections before safety was affected. Only a small proportion of all damage has led to leakage or other safety deficiencies as a result of undiscovered cracking and other degradation. On the other hand, some operating conditions are changed in connection with implemented and planned power uprates, as well as planned operation exceeding the period of operation of approximately 40 years that the facilities were originally designed for.

An analysis of the information contained in 'STRYK', SSM's damage database for operation induced damage in passive mechanical components reported from Swedish nuclear power plants, shows that some of the most commonly occurring factors causing damage are represented by flow accelerated corrosion (FAC), intergranular stress corrosion cracking (IGSCC), transgranular stress corrosion cracking (TGSCC), in addition to intergranular stress corrosion cracking in primary systems of pressurised water reactors (PWSCC).

Guides are available for the mechanical properties of different kinds of rubber and other polymer materials and their temperature stability, oxidation stability, water absorption and resistance to organic compounds. What is less known is how these properties can be sustained over a long-term sequence. The radiation stability of polymer materials is also relatively unknown.

Measures to be taken for long-term operation

There is generally a good level of knowledge about most of the factors that have an impact on the damage mechanisms that are presently known. These contributory factors are also reasonably taken into account as a basis for the programmes for water chemistry follow-ups and inspections applied now, both in terms of the scope and intervals of inspections and control. As far as concerns some mechanisms that can cause damage to metal materials, however, the present situation in terms of technical expertise is not yet fully sufficient for application of effective programmes for water chemistry follow-ups and inspections. This situation mainly applies to a long-term perspective. Considering the longer periods of operation, the inspection programmes may need to be adjusted while taking into account new research findings on:

- IGSCC in the nuclear grade material that has been used to replace damaged components and, for preventive purposes, also components manufactured of materials susceptible to damage
- PWSCC in pressurised water reactor components in general, and also for the materials of the Alloy 52 and Alloy 690 types used to replace damaged components and, for preventive purposes, also components manufactured of materials susceptible to damage
- IASCC, that is, irradiation assisted stress corrosion cracking in reactor pressure vessels' internal components, both in terms of initiation and spreading of cracks
- TGSCC, particularly in terms of crack initiation

Experience has shown that damage mechanisms can arise in components where metal materials and environments were previously of kinds assessed as being resistant to degradation. One example is environmental cracking in some materials where the initiation period is very long. This is why a major focus needs to be placed on these phenomena in connection with extended periods of operation and applying inspection programmes that as far as reasonably possible are also capable of detecting completely unknown damage mechanisms and also known damage mechanisms that manifest themselves in unexpected places. Continued research is also needed here into (among other things) the cracking susceptibility of metal materials in different environments.

As far as concerns the FAC mechanism, current knowledge about contributory factors is sufficient, but special attention may need to be given to the scope and orientation of inspection programmes at facilities where the operating conditions are changed through power uprates and modernisation work. When it comes to facility management of FAC, a combination of analysis programmes, experience-based assessments and information from facility databases should be suitable.

For the mechanism of boric acid corrosion, current knowledge is, in terms of contributory factors, also assessed as adequate, but taking into consideration international damage experience demonstrating the potentially aggressive nature of this mechanism, it is important that the programmes for maintenance and monitoring at pressurised water reactors take particular account of signs of leaking borated water.

Research has focused especially on environmental tolerance and environmental qualification of electrical components, but these efforts must now be expanded to also encompass other polymer materials used in reactor containments. This is a prerequisite for developing and maintaining effective environmental qualification and ageing management programmes for these polymer materials.

Beyond this, the facilities should check whether polymer construction materials also perform safety functions at other places in the nuclear power plants, for example, in components in close proximity to reactor containments, in coolant channels and heat exchangers as well as in penetrations not classified as electrical components. Rubber-lined pipes, filter frames and ion-exchange resins can also be mentioned in this context.

There are more non-metal materials at nuclear power plants that are subject to ageing, for example kinds of plastic, adhesive sealants, elastomers with fillers such as reinforcements or flame retardants, paint, coatings, finishes, epoxy resins, oils, lubricants, etc. in addition to inorganic materials. This is why the licensees must perform systematic inventories of these materials in the facilities, both in terms of safety function and occurrence, in addition to subsequent additions to the ageing management programmes as necessary.

3.4. A few particular ageing issues and aspects that should be taken into account prior to long-term operation

3.4.1. Irradiation embrittlement of reactor vessels

The mechanism and current knowledge

The function of the reactor pressure vessel includes enclosing and supporting the reactor core and the reactor's internal components. The reactor pressure vessel of a boiling water reactor (BWR) is around 20 metres tall, approx. 6 metres in diameter and has a thickness of approx. 15-20 cm. The reactor pressure vessel of a pressurised water reactor (PWR) is slightly smaller: approx. 13 metres tall, approx. 4 metres in diameter and having a thickness of approx. 20 cm. The material in the pressure vessels is a low-alloy quenched and tempered steel with a carbon content of a maximum of 0.25%. It is a molybdenum and nickel alloy but also contains low levels of chromium, copper and vanadium. With small variations, all Swedish reactor pressure vessels largely have the same material qualities.

Since the reactor pressure vessel contains the reactor core, it will be exposed to neutron irradiation. This irradiation affects the material over time so that the brittle transition temperature, the transition between brittle and ductile, increases and the impact energy in the ductile area drops at the same time as the yield strength increases. Since fast neutrons (E > 1MeV) are needed to embrittle the pressure vessel material, only the parts of the pressure vessel that are close to the reactor core (belt line) are embrittled. Other parts of the pressure vessel, above and below the reactor core, are not affected by the neutron irradiation.

What basically happens in the material is that the neutrons collide and displace atoms from their regular lattice sites within the metals. This creates vacancies and interstitials in the material's structure. Early research findings identified the fact that copper atoms easily form precipitations in reactor vessel steel during irradiation of fast neutrons. The copper precipitations give a particle hardening that is the main cause behind embrittlement of the steel. It has gradually been demonstrated that other alloy materials such as nickel, phosphorus, manganese and, to a certain extent, silicon, contribute to the embrittlement. The effect of these alloy elements emerges only after a relatively long period of time and in such a way so that the rate of embrittlement increases substantially after a certain period of time in operation. This phenomenon is called 'late blooming phases' (LBP). It is still unclear whether LBP presuppose the presence of copper precipitations. These mechanisms are complicated and the links are not fully understood. Embrittlement also depends on temperature, so that irradiation at a high temperature gives a lower degree of embrittlement compared with irradiation at a low temperature.

Designing and dimensioning of reactor pressure vessels involves analysis of the degree of embrittlement of the material over its entire intended lifetime. The design is dimensioned so that it can withstand postulated defects with an embrittled material at its end of life, or 'EOL'. Verifying these analyses is performed under a 'surveillance programme' where specimens manufactured of the pressure vessel material in question and the relevant weld material is placed in the pressure vessel between the reactor core and the reactor pressure vessel. There, specimens are exposed to somewhat higher neutron irradiation, fluence, compared with the reactor pressure vessel. The calculated degree of embrittlement can then be checked by removing and testing specimens at predetermined points in time. All Swedish reactor pressure vessels have surveillance programmes[9] in accordance with the provisions of SSM's regulations. Removal and testing of irradiated specimens have been performed at least once for all reactor pressure vessels.

Preconditions for long-term operation include a thorough analysis of surveillance data and the reactor pressure vessel's degree of embrittlement. The surveillance programme must be revised and adapted to the extended period of operation. This is why all Swedish reactors are now subject to investigations into extended periods of operation that also encompass updated analyses of reactor vessel fluence in connection with EOL. It should be mentioned in this context that the increased fluence implied by a power uprate results in an update of the analyses of maximum permissible limiting value of the pressure and temperature (HTG), which are reviewed by SSM.

The degree of irradiation for boiling water reactors is low, even for 60 or 80 years of operation. No substantial problems with embrittlement of reactor pressure vessel materials

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[9] Ringhals 3 and Ringhals 4 share the same surveillance programme.

are foreseen provided that no new experiences emerge indicating presently unknown phenomena.

The average fluence for pressurised water reactors is nearly 50 times higher. There is currently awareness of relatively severe irradiation embrittlement of welds having a high level of nickel content at Ringhals 3 and Ringhals 4. When these pressure vessels were manufactured, welds with a high level of nickel content were intentionally chosen in order to produce ductile welds having a low brittle transition temperature. At this point in time, the effect of nickel on embrittlement of reactor pressure vessel material was not known, which is why the choice of weld was correct in accordance with the level of knowledge at the time. Ringhals AB is closely monitoring the embrittlement situation and has taken measures to mitigate the degree of ongoing embrittlement.

One problem related to long-term operation regardless of reactor type is that a shortage of specimens for surveillance testing can arise. Most of the reactors have extra specimen chains that can be used for extended periods of operation, and for some of them, replanning of specimen removal is possible so that the testing programme can be adapted for an extended period of operation. However, it may be relevant in some cases to reuse specimens. Here, methods have been subjected to testing and applied on the part of reactors abroad.

Measures to be taken for long-term operation

Against this background, SSM’s view is that for periods of operation exceeding approximately 40 years, licensees will need to:

- revise and adapt the programme for surveillance testing to the extended period of operation, among other things owing to the long-term impact of certain alloy materials on embrittlement (late blooming phase) as well as the fact that a shortage of specimens might arise
- perform time-dependent analyses of the reactor pressure vessel’s degree of embrittlement and its significance for structural integrity, where such analyses need to include analyses of maximum permissible limiting value of the pressure and temperature (HTG), an estimate of upper shelf toughness, acceptable crack sizes in the belt line region as well as the frequency of leakage and breaks in the belt line region

These programmes and analyses will also serve as necessary input for SSM’s decision-making when it comes to long-term operation of a nuclear power plant. For more information about SSM’s point of view on the analyses that need to be performed, please refer to report [13].

SSM will monitor and support research projects on embrittlement mechanisms when it comes to reactor pressure vessel materials and extended periods of operation. The Authority also intends to revise and clarify its regulations and general advice concerning applicable requirements and assumptions of reactor pressure vessels’ degree of embrittlement and its significance in terms of structural integrity. For more information about this need to clarify requirements and general advice, see the relevant report [3].

3.4.2. Fatigue

Rules applied, current knowledge and concerns

As a rule, dimensioning against component fatigue at nuclear power plants takes place during the design stage, and in most cases, by applying the American pressure vessel standards contained in Section III of early editions (prior to the 2010 edition) of the ASME Boiler and Pressure Vessel Code. This implies use of the budgeted and expected number of transients for calculation of accumulated usage factor U that is determined by the ratio between the number of cycles and the permitted number of cycles, whereupon U is evaluated for each expected load event. It is required that $U \leq 1.0$. The permitted number of cycles is determined by the experimentally determined 'design curves' where the stress amplitude as a function of the number of cycles is presented. The design curves already take into account a safety margin added to the mean curve used during the fatigue experiments performed on small polished specimens in an air environment. The safety margin is to cover the spread in material data, the difference between laboratory specimens and actual components, in addition to effects from surface roughness. However, it has been noted that ASME III particularly points out that the experiments on which the design curves are based do not include the presence of corrosive environments that can shorten the fatigue life.

Recent years' research conducted in Japan and the United States, for example, has indicated that the mean curves for fatigue on which the design curves for austenitic stainless steel in the early editions of ASME III are based, are non-conservative in the area number of cycles exceeding 10^4. New experimental data from the reactor water environment on the part of ferritic steel, austenitic stainless steel as well as nickel-base alloys has also indicated that environmental effects can reduce the fatigue life significantly compared with an air environment.

This implies a concern that reactor facilities in Sweden (that have been dimensioned against fatigue in accordance with the early rules of ASME III but without taking explicit account of the reactor water environment) will eventually develop fatigue cracks in areas that were unanticipated when the reactors were designed. Consequently, there is not always an inspection programme in place for this for detection of possible damage at an early stage. Even if fatigue cracks do not arise, the actual safety margin may prove to be significantly smaller than originally intended.

The licensees have stated that today's inspection programmes detect possible damage that does arise as a result of low-cycle fatigue. However, there is the possibility that an area where damage index = III, for which $U < 0.3$, is assigned inspection group[10] C if consequence index = 2 (which e.g. applies to the feed water system inside the reactor containment), but would, using an updated fatigue analysis, receive $U > 0.7$ whereby damage index = I would give inspection group A. This means that there may be areas not identified in current inspection programmes, and which prior to extended periods of operation merit greater focus in terms of potential fatigue damage. This mainly applies to areas in the primary system, but also certain internal parts subject to fatigue, such as the moderator shroud support of boiling water reactors. This concern mainly relates to periods of operation exceeding 40 years. Nothing dramatic is anticipated to occur at 40 years of operation, though after this point, the risk of undiscovered fatigue cracks may gradually rise.

It is important to keep in mind that low-cycle fatigue is a slow process and that today's inspection programmes have been designed to detect potential damage to the systems most subject to fatigue. This can give an early warning as to whether low-cycle fatigue is an

[10] Mechanical components are to be classified into inspection groups A, B and C defining the scope and orientation of in-service inspections. As far as concerns boiling water reactors, a classification system is currently used with a damage index and consequence index, which govern the parameters of the inspection groups.

active damage mechanism that can justify a reassessment of whether additional areas with lower usage factors need to be encompassed by the inspection programme. On the other hand, there is no guarantee that the areas pointed out today as having the highest usage factors will remain as such considering environmental effects, in other words, the question of whether the ranking between usage factors will be the same considering the environmental effects.

Measures to be taken for long-term operation

Against this background, the SSM view is that a licensee, prior to extended periods of operation, should demonstrate through analyses that no unacceptable risk of fatigue cracking will arise up until the end of the analysis period as far as concerns areas of the reactor facility sensitive to fatigue. These analyses need to take into consideration the impact of the reactor water environment. For more information about SSM's point of view on the analyses that need to be performed, please refer to the reports [3] and [13].

As a basis for these analyses, the licensee should compile data on the accumulated number of transients occurring in the reactor in addition to the expected number of transients up until the end of the analysis period for subsequent comparison against the transient budget serving as the basis of the reactor's design. In this context, it should be particularly taken into account whether the transients that have occurred are assessed as worse (e.g. occurring faster or with a broader range of temperature) than for the transient budget in the design assumptions.

SSM is also of the view that the present system for recording and evaluating transients needs to be examined. It is appropriate to have some kind of automatic recording of transients that have occurred, as well as defined measurement points and criteria defining the startup and shutdown conditions for the transients.

The results of these follow-ups and analyses will also serve as necessary input for SSM's decision-making when it comes to long-term operation of a nuclear power plant.

SSM intends to revise its regulations concerning mechanical components[11] in terms of the applicable requirements and assumptions as far as concerns reviewing a reactor's degree of fatigue for long-term operation as well as reporting of the number of accumulated transients in the reactor.

3.4.3. Thermal ageing of cast stainless steel and welds

The mechanism and current knowledge

Thermal ageing is defined as a change, dependent on time and temperature, of the microstructure of a material that leads to reduced ductility and deterioration in its fracture toughness properties. The material becomes more brittle over time. Usually, this change to the microstructure also gives the material a greater strength; that is, greater yield strength, rupture strength and hardness. When it comes to boiling water reactors and pressurised water reactors, the main materials that are sensitive to thermal ageing are austenitic cast stainless steel, stainless welds, martensitic stainless steels and precipitation hardened martensitic stainless steels. Austenitic stainless steel, manufactured by forging or rolling, and nickel-based material, are not sensitive to this ageing mechanism.

[11] Presently SSMFS 2008:13

Stainless welds and cast stainless steel are often used for pressure-bearing components, for example welding of pipe systems and for valve and pump casings. Martensitic stainless steels and precipitation hardened martensitic stainless steels are used for load-bearing details, such as internal parts of valves and pumps. Consequently, for safety reasons, the damage mechanism of thermal ageing is primarily relevant for stainless welds and cast stainless steel. Martensitic materials must be kept in mind; however, the effect of thermal ageing is smaller as far as these are concerned. Also, these components are usually relatively straightforward to replace.

Embrittlement due to thermal ageing is a result of three mechanisms: a) formation of a chromium enriched α′ phase through spinodal decomposition of ferrite, b) precipitations of the G phase in ferrite, and c) the formation of carbides in the phase boundary between austenite and ferrite. Of these three, the formation of the α′ phase is the main mechanism behind thermal ageing of cast stainless steel and stainless welds at operating temperatures for boiling water and pressurised water reactors. The third mechanism, the formation of carbides, is not active at temperatures below approx. 425°C and is thus not an active degradation mechanism during normal operation. As far as concerns the second mechanism, precipitations of the G phase in ferrite, it has not been clearly established to what extent it contributes to embrittlement. Tests have shown that if aged and brittle material is heat treated so that the α′ phase is dissolved but the G phase is unaffected, the original level of fracture toughness properties is restored. This area requires further research in order to understand the kinetics and interrelationship between the different mechanisms.

At an approximate 286 °C operating temperature of boiling water reactors, the process is much slower than compared with an approximate operating temperature of 325 °C in pressurised water reactors. The studies made into austenitic cast stainless steel and stainless welds indicate that harmful embrittlement at Swedish nuclear power plants due to thermal ageing is unlikely in connection with periods of operation just over 40 years. In many cases, studies indicate that longer periods of operation are fully acceptable. Consequently, the damage mechanism of thermal ageing does not become a design limiting assumption when a facility is approaching, or is passing, 40 years of operation.

Internationally, however, the topic of discussion today is 60 years of operation or longer. From this perspective then, additional studies and further research need to be conducted in the area of thermal ageing.

Measures to be taken for long-term operation

Against this background, the SSM view is that harmful embrittlement due to thermal ageing at Swedish nuclear power plants is unlikely in connection with periods of operation of around 40 years. For even longer periods of operation (around 60 years), additional studies and further research need to be conducted in the area of thermal ageing.

3.4.4. Concrete constructions and reactor containments

General information about concrete constructions at nuclear power plants

Similar to other materials, concrete breaks down over an extended period of time. The rate of this process depends on the environment and loads that the concrete is exposed to.

Loads give rise to time-dependent deformations and the impact from the environment brings about changes to the concrete's properties over time. Concrete consists of aggregates stabilized by cement paste (cement and water). The proportion of water in relation to cement, what is known as the water-cement ratio, is important when it comes to the structural integrity and durability of concrete. Generally, the level of structural integrity and durability rises with a lower water-cement ratio. The kind of cement and the ballast's mechanical properties also have an impact on the concrete's structural integrity and durability.

A durable concrete construction presupposes that it has the correct composition and has been adapted for its environment. Among other things, the ballast must be adapted to the kind of cement and the cement mixture must have been adapted to the chemicals present in the environment. What's more, the concrete needs to be free of cracking, or alternatively, that existing cracks do not serve as channels transporting corrosive substances that degrade the concrete.

From the point of view of nuclear safety and radiation protection, the reactor containment is the most important concrete construction in a nuclear power plant. In principle, all containment constructions in Sweden have the same design: protected on the inside by a cylinder-shaped concrete wall and a containment steel liner, with the exterior surrounded by a load-bearing and cylinder-shaped concrete construction. These three parts of the construction join together physically to form a unit and help to meet the performance requirements imposed. The prestressed steel reinforcement represents the most important design component for the containments' capability to perform their function. All containments also contain ordinary non-prestressed reinforcement.

The internal diameter of reactor containments is between 19 and 25 metres. The prestressed reinforcement steel is placed both horizontally and vertically in the cylindrical part of the containment. Pressurised water reactors also have prestressed steel reinforcement in the containment dome. The prestressed cables are often placed in casing tubes. At Ringhals 2, 3 and 4 and Forsmark 1, 2 and 3, the casing tubes have either been injected with corrosion protectant such as grease, or they are ventilated with dry air. Their steel cables (non-grouted tendons) are accessible for inspections and replacements. The casing tubes of Oskarshamn 1, 2 and 3 and Ringhals 1 (previously also Barsebäck 1 and 2) containing steel cables have been injected with cement (grouted tendons). For this reason, these prestressed cables are not accessible for inspections nor replacements in the same way as the non-grouted tendons. Ordinary non-prestressed reinforcement is used in addition to prestressed reinforcement steel.

The cylindrical wall of the reactor containment is cast in two concentric parts having a total thickness of 1.0-1.5 m, with the inner concrete part's thickness being approx. 0.2-0.3 m. There is a cast containment steel liner between the two concentric parts that is 4-8 mm thick. In some cases, the containment steel liner is manufactured of stainless steel material. The containment steel liner of the boiling water reactors' baseplate is freely exposed in the bottom of the condensation pool. At Ringhals 1 and Oskarshamn 2, the containment steel liners in the upper parts of the containment are freely exposed. In the spherical dome of the pressurised water reactors, the containment steel liner is freely exposed.

Breakdown and degradation mechanisms

Concrete constructions degrade over time from not only operation, but also functional degradation, as well as due to different environmental factors (environmental degradation). Functional degradation can be caused by different kinds of static and dynamic loads, changes in temperature, shrinkage and creep. Environmental degradation can be caused by external degradation substances or by the material itself. When a reinforced concrete construction degrades, this is usually the result of several interacting mechanisms. One degradation mechanism helps to activate other degradation mechanisms.

The loads that the concrete construction is designed to bear also have an impact on the construction's properties. Reactor containments are less affected by normal operating loads since they have been dimensioned for much larger loads. Degradation of reinforced concrete constructions due to environmental factors can be divided into four main categories:

- Physical degradation
- Chemical degradation
- Corrosion (containment steel liner, steel reinforcement bars and prestressed steel)
- Radiation degradation

Concrete that has no reinforcement nor embedded steel details sensitive to corrosion is very resistant against environmental degradation. A common problem area in connection with environmental degradation is a lack of interaction between concrete, reinforcement and other embedded materials.

The damage and deterioration occurring over the years show that they are mainly caused by deficiencies in connection with construction or later plant modifications. These deficiencies are of a kind that deviates from design drawings or from the design basis. This kind of damage has for example been observed at Barsebäck 2, Forsmark 1, Oskarshamn 1, Ringhals 1 and Ringhals 2 and mainly constitutes corrosion damage in the metal parts of containments. Similar experience has been gained internationally.

Inspections and tests of containments

The main way of checking reactor containments is through leak tightness testing. This is to check containment steel liner integrity and leak tightness as well as the leak tightness of air locks, penetrations and isolation valves. These tests are initially conducted in connection with a static gas pressure corresponding to an estimated maximum pressure during a design basis accident. Subsequent testing normally takes place at a 50 per cent level of this pressure. At some of the facilities, a higher level of pressure is used. Both gas and water leakage through isolation valves plus penetrations are measured. Containments are leak tested three times at more or less regular intervals over a ten-year period.

As with other pressure and leak tests, however, leak tests of reactor containments only illustrate the situation at a given moment in time. Emerging degradation is normally not discovered by this kind of testing. Early indications of initial degradation require application of other kinds of non-destructive testing (NDT). Non-destructive testing is only used to a very limited extent today, but it increases in relevance as the facilities age. At the same time, however, this kind of method cannot be used for testing of the large areas and volumes that make up a reactor containment. This is why investigations and research are being conducted nationally and internationally to develop methods for the identification of parts of a reactor containment that are known to be sensitive to degradation, in addition to non-destructive testing methods for inspecting such parts. These investigations and research projects have been evaluated and compiled in report [3]. This compilation for instance shows that today's NDT methods are used to:

- verify the concrete construction's agreement with drawings
- detect cracks and to some extent determine their size on concrete surfaces

- detect significant corrosion on containment steel liners assuming reasonable accessibility for testing
- investigate corrosion on ordinary/non pre-stressed reinforcements

There is limited potential for using NDT when it comes to grouted tendons. In favourable conditions, it is possible to identify tendon breaks using NDT. There is negligible, or very little, potential for identifying corrosion or other damage to grouted tendons. It is important to intensify efforts to develop verified and validated analysis methods for reliably predicting loss of tendon prestress forces. This also applies when developing methods for checking loss of tendon prestress forces and the presence of corrosion damage.

Measures to be taken for long-term operation

Against the background described above, the SSM view is that for periods of operation exceeding approximately 40 years, licensees will need to:

- verify the as built concrete constructions against the original underlying drawings for identification of any deviations that might lead to degradation, e.g. corrosion, and develop an action plan when such deviations are discovered
- conduct analyses for non-grouted tendons in the reactor containment that demonstrate variations in tensioning force as a function over time
- develop analysis methods and NDT methods for more general checks of the status of grouted tendons in the reactor containment
- identify critical areas in terms of reactor containment leak tightness and load bearing capacity during various load conditions and develop suitable NDT methods for more extensive condition monitoring

The results of these verifications, analyses, investigations and testing will also serve as necessary input for SSM's decision-making when it comes to long-term operation of a nuclear power plant. For further information about SSM's point of view on the areas needing particular attention prior to long-term operation, please refer to report [3]. For more information about the time-dependent analyses that need to be performed, please refer to report [13].

SSM intends to draft regulations concerning concrete constructions with a particular focus on reactor containments. The regulations need to impose requirements covering materials, structural integrity analyses and inspection methods for timely discovery of potential ageing phenomena. This also applies to ageing of tendons and embedded steel liners in reactor containments.

Further research and development is needed in the area, e.g. in terms of:

- irradiation effects on concrete in certain vulnerable parts
- impact from currents in earthing grids and electromagnetically induced currents on the corrosion process for steel constructions embedded in concrete
- methods for checking loss of tendon prestress forces in grouted tendons
- models for predicting dehydration and shrinkage of reactor containment concrete and their impact on loss of tendon prestress forces in grouted tendons
- methods for checking thick concrete components using NDT

3.4.5. Electrical and instrumentation & control (I&C) equipment

Current knowledge in terms of physical ageing
Swedish nuclear power plants' safety systems and their respective I&C equipment are nearly completely dependent on a supply of power. These safety systems are also very dependent on the actual performance of the I&C equipment. Electrical equipment (primarily cables) is a key component of a facility's power supply to its safety systems. Cables are also components of crucial importance in connection with signal transmission as part of the safety systems.

Faulty electrical and I&C equipment can have an immediate impact on safety; however, it is easier to rectify most problems, if they are known, compared with many mechanical components and building structures.

Generally, the level of knowledge is good about ageing phenomena and ageing management on the part of electrical and I&C equipment in the nuclear power industry. What's more, national and international programmes for research and exchange of experience are in progress within the framework of the IAEA and OECD/NEA for the purpose of continually raising the level of knowledge about extending existing nuclear power plants' periods of operation.

Physical ageing and environmental qualification of electrical components for withstanding harsh environments have been dealt with over a long period of time, mainly when it comes to cables. There is a potential risk that physical ageing can result in simultaneous failure of components of the same kind in redundant systems (common cause failure, or 'CCF'). This consequently presupposes identification and increased alertness to the possible occurrence of this potential risk. Environmental qualification of components in terms of physical ageing has the purpose of ensuring a qualified life for a certain service environment and given and specified performance. All Swedish licensees have an environmental qualification programme.

SSM (and its predecessor, SKI) was early when it came to limiting the parameters for the assessments on qualified life based upon initial environmental testing prior to installation. The strategies presently applied in Sweden may also be applicable to long-term operation. An updated review may nevertheless be needed to ensure correct management and follow-ups during long-term operation.

When the qualified life has ended, then the components will need to be replaced by new qualified replacement components for the remaining period of time. Components meant to be retained need to have a certain proven remaining life endurance, which presupposes updated qualification. It is often a financial consideration when it comes to changing to new qualified components or updating the qualification as some cables are very difficult to access for a replacement, or there are large quantities of components needing replacement.

In most countries, awareness of the need for different kinds of follow-ups of component condition after installation has grown. Experience-based knowledge demonstrates that individual methods for condition-based testing have their limitations, which is why a combination of condition-based testing should be used to counteract the individual methods' limitations. It is likely that more research is needed here to achieve an optimal combination method. Several studies and recommendations in other countries reinforce the belief that increased monitoring of components' operating environment should take place prior to long-term operation. This is a precondition to ensure that the surrounding environment is not more severe than defined in previous environmental qualifications.

Current knowledge in terms of technological ageing

Technological ageing (phasing out equipment) is receiving more attention. This is why these aspects also need to be taken into account in the licensees' ageing management programmes and ageing management activities. This is particularly important when it comes to I&C equipment.

A fairly common belief in Sweden is that a shortage of the right spare parts or service mainly has an impact on production and not safety since the Operational Limits and Conditions ('STF') specified will end operation unless all the operability criteria have been fulfilled. On the other hand, spare parts that have long been in storage and whose ageing status is unknown pose a potential safety risk. Here, the storage environment also serves as a factor in terms of ageing spare parts. Another problem that has been pointed out recently is the fact that replacement spare parts do not have the same robustness nor the same properties as original spare parts. To date, however, these cases have not applied to electrical and I&C equipment in the nuclear power industry. An additional problem that has emerged when it comes to replacement spare parts is 'counterfeit' products that can be difficult to identify since even the various proof of origin documents look real but are fake.

Prior to and while replacing ageing equipment, new problems arise that can have an impact on safety. New equipment is often based on a different technology, can behave in a different way in abnormal situations than compared with the original equipment, or complete knowledge about the new equipment has not reached the user. The Forsmark incident in 2006 demonstrated this, for example the plant's new generator protection and its new equipment for an uninterruptible power supply to the safety systems.

This also relates to design process compliance and implementation, which are key to safety. Since electrical and I&C equipment are changing while also becoming more complex, and also that the personnel who are familiar with the older systems are ageing and entering retirement, design and verification work is becoming more difficult. This may provide justification for looking into whether updated or changed design and verification processes might be a way of ensuring a high level of safety for the new and increasingly complex and software-based electrical and I&C systems being introduced in conjunction with modernisations.

Measures to be taken for long-term operation

Against the background described above, the SSM view is that for periods of operation exceeding approximately 40 years, licensees will need to:

- Introduce increased monitoring of components' surrounding environment over the extended period of operation. This is a precondition to ensure that the surrounding environment is not more severe than assumed in previous environmental qualifications. This is especially important after power uprates.
- Review environmental qualification programmes to ensure correct management and follow-ups during long-term operation. This for example applies to accelerated ageing. Applying condition-based qualification as a complement should also be considered.
- Replace equipment with an expired environmental qualification period with new qualified replacement equipment for the remaining period of operation, or, alternatively, demonstrate by means of updated qualifications that existing

equipment can be used throughout the entire, or parts of, the extended period of operation
- Review the ageing management programme so that it also covers all aspects of technological ageing of electrical and I&C equipment
- Review the management of spare parts and the storage environment so that spare parts that have been kept in storage for a long time do not pose a potential safety risk
- Introduce routines to ensure that new spare parts for electrical and I&C equipment have the same properties and quality as the original parts

Descriptions of these programmes, overhauls and other measures will serve as necessary input for SSM's decision-making when it comes to long-term operation of a nuclear power plant. For more information about SSM's point of view on measures needing to be taken prior to and during long-term operation, please refer to report [3].

SSM also intends to revise its regulations and general advice within the area to ensure that they become more comprehensive, also so that they cover analyses and measures needed for long-term operation.

3.5. Periodic safety reviews for long-term operation

3.5.1. Periodic safety reviews

Requirements on integrated and periodic reviews of safety at Swedish nuclear power reactors have been in force since the early 1980s. Through Government Bill 1980/81:90 concerning guidelines for energy policy, the Swedish Government obtained the Riksdag's approval of certain guidelines applying to safety for operating nuclear power plants. These guidelines implied that each nuclear power plant would, during its technical lifetime, undergo "no less than three complete safety reviews similar to the one preceding the granting of the licence for starting up a reactor facility for the very first time". This kind of review, an 'As operated Safety Analysis Report', or ASAR, was to, under these guidelines, take place every eighth to tenth year. The aim of an ASAR was to provide impetus for safety improvements against the background of new knowledge, technology and methods. ASARs were to signify a check of safety at a nuclear power plant.

Corresponding requirements, that nowadays are more forward-looking, are also imposed by many other countries; also, the International Atomic Energy Agency (IAEA) has issued recommendations for this kind of periodic safety evaluation. Internationally, these are called 'Periodic Safety Reviews', or PSRs.

The requirement on implementing periodic reviews of safety and radiation protection was subsequently incorporated first in SKI's and later in SSM's regulations. Since 2010, these requirements are part of the Act on Nuclear Activities (1984:3). Section 10a states that a party that holds a licence to possess or operate a nuclear facility shall, at least every ten years, conduct an overall assessment of the facility's safety and radiation protection. The assessment shall be conducted in relation to developments in science and technology. It shall include analyses and descriptions of:
- the way in which the facility's design, function, organisation and operations fulfil the requirements imposed by the Act on Nuclear Activities, the Environmental

Code and the Radiation Protection Act in addition to regulations and conditions issued under this legislation, and
- the prerequisites for compliance with these regulations and conditions up to the next overall assessment.

It is also stipulated that the overall assessment and the measures ensuing from it are to be reported to SSM.

It is (among other things) pointed out in Government Bill 2009/10:172 [12] that periodic safety reviews are an important and fundamental principle for safety and radiation protection work at nuclear facilities. Periodic safety reviews are an effective instrument of regulatory supervision whose aim is to gain an overall understanding of safety at a facility to make it possible to determine reasonable and feasible measures for maintaining a high level of safety. When it comes to ageing reactors, this is also a matter of improving safety so that they as far as possible are as safe as relatively new reactors. It is also pointed out in the Government Bill that periodic safety reviews of nuclear power reactors also provide safety-related indicators that, over a long-term perspective, might have a negative impact on the national power supply.

The periodic safety review is to cover analyses and descriptions of how a facility's design, function, organisation and operations fulfil the conditions and regulations issued under the Act on Nuclear Activities and Radiation Protection Act as well as how the facility fulfils the requirements imposed by the Swedish Environmental Code, above all the general rules of consideration contained in Chapter 2. It is pointed out in the Government Bill that the periodic safety review thus signifies a process that enables the supervisory authority to gradually impose more stringent safety requirements in connection with nuclear power plant operation. Periodic safety reviews are to be conducted at least every ten years. Periodic safety reviews may take place at more frequent intervals if justified by safety-related indications.

The intentions of the legislator are consequently well in line with developments internationally when it comes to the use of PSRs as a key instrument of regulatory supervision for the authorities as part of their task to ensure that licensees gradually improve the facilities' safety while taking into account operating experience, insights gained from safety analyses, technological progress and findings from safety research.

In June 2010, the CNRA set up its Senior Task Group with the aim of developing recommendations for the regulatory authorities' work on adopting a position on, and supervision of, what is referred to as 'long-term operation'. In its final report, the Group pointed out the need for clear and formal processes, both for adopting a position on long-term operation as well as for regulatory supervision over the extended period of operation. Two different ways are recommended: [1)] formal renewal of the operating permit (licence renewal), and [2)] adopting a position within the framework of a periodic safety review (PSR). This is because of differences in national legislation between countries and whether or not the permits/licences are limited in time. In the United States, licences are usually limited to 40 years of operation for reactor facilities, whereas in many European countries, licences are in effect as long as safety requirements are fulfilled.

According to both the recommended ways, facilities' ageing management and ageing management programmes are of key importance when adopting a position on long-term operation. In countries where PSRs are applied, it is also recommended that safety upgrades and comparisons against new safety standards should serve as key elements of the regulatory authorities' review work and their adopting a position on long-term operation. See also Chapter 4.

3.5.2. Adopting a position on long-term operation

As licences for Swedish nuclear power plants are not limited in time, SSM shares the CNRA's point of view that it is appropriate to adopt a position on long-term operation on the basis of a periodic safety review.

Updates are currently in progress of SSM's steering documents covering examination of licensees' periodic safety reviews. SSM also intends to clarify and specify more precisely the requirements on the periodic safety reviews, but has already established that an account of a licensee's periodic safety review that is to serve as the basis of the Authority's position on long-term operation needs to (for example) encompass the results from time limiting ageing analyses (TLAA) and other information about measures prior to long-term operation as described in sections 3.3 to 3.4 above. For more information on SSM's point of view on TLAA for mechanical components, please see report [13].

3.6. Conclusions

Many of the damage and ageing mechanisms that occur, or might occur, at Swedish nuclear power plants can be dealt with satisfactorily with the inspection programmes and ageing management programmes applied today. Thus, these programmes should give good preconditions for safe operation, also in connection with long-term operation. There are, however, a number of areas in which ongoing inspections and analyses, in addition to development of methods and knowledge, are prerequisites so that these programmes can more effectively detect early indications suggesting safety deficiencies due to ageing over extended periods of operation. These programmes also need to be designed so that they as far as reasonably possible are capable of detecting completely unknown damage mechanisms and also known damage mechanisms that manifest themselves in unexpected places.

In connection with power uprates, which mainly imply an increased rate of flow in certain systems in addition to higher neutron doses, it is important to be alert to an elevated risk of flow assisted corrosion (FAC) and vibrations, in addition to an elevated risk of irradiation embrittlement of reactor vessels and irradiation assisted stress corrosion cracking (IASCC) of internal parts. As far as concerns environmentally qualified electrical, instrumentation and control equipment, a greater focus should be placed on e.g. components' surrounding environment after power uprates so that the environment is not more severe than assumed in the original environmental qualifications.

What's more, prior to and during long-term operation, special attention also needs to be given to:

- irradiation embrittlement of reactor pressure vessels, particularly taking into account effects that can substantially increase the rate of embrittlement
- fatigue, particularly taking into account impact from the reactor water environment on areas sensitive to fatigue
- the condition of tendons and steel liners in reactor containments
- degradation mechanisms that can influence reactor containments' concrete and metal parts

- possibilities for reliable inspections and testing of reactor containments
- the validity of environmental qualification of electrical, instrumentation and control equipment as well as other systems containing construction materials made of polymers

Ongoing knowledge building is necessary for the long-term application of effective inspection programmes in terms of stress corrosion in: a.) pieces of equipment manufactured of nuclear grade material, and b.) pieces of equipment in pressurised water reactor environments.

As far as concerns even longer periods of operation approaching 60 years, continued investigation and research are needed for timely detection of degradation due to loss of fracture toughness impact strength as a consequence of thermal ageing of stainless steel welds and cast stainless steel.

Thus, what is crucial as to whether a reactor can be operated further over extended periods and with a sustained level of safety is the licensee applying a thorough and effective ageing management programme, also where a licensee's ageing management activities encompass:

- excellent knowledge about factors affecting ageing on the part of all inherent systems, structures and components of importance for safety at a nuclear power plant
- sufficient input data encompassing all necessary information about all systems, structures and components (e.g. inherent materials and environmental conditions) needed for assessing their ageing status
- effective programmes for preventive and corrective maintenance
- effective programmes for regular testing and monitoring of all critical parts of the facility so that signs of ageing damage are discovered in time
- an action plan that gives a detailed account of the measures to be taken in connection with signs of ageing damage so that safety requirements are met
- information about the extent of degradation that is acceptable
- good experience feedback and a proactive approach implying that measures are taken continuously in pace with available knowledge about how to prevent ageing damage

A licensee must also have:

- an effective organisation and a suitable management system that controls and leads ageing management activities while also ensuring that personnel with sufficient professional skills are available now and in the future, and that enables identification of problems related to ageing and their being dealt with in a way so that safety requirements are met
- a research programme that develops new knowledge about ageing phenomena including contributory factors and models, development of analysis methods, development of effective inspection methods and measures to mitigate ageing in addition to these results being implemented continuously in the ageing management programmes

On the basis of a periodic safety review, SSM intends to adopt a standpoint on long-term operation under the requirements imposed by the Act on Nuclear Activities and the Authority's regulations. Clarification and more precise wording of SSM's regulations and general advice concerning periodic safety reviews are being planned in terms of aspects of importance in connection with long-term operation. SSM has nevertheless already established that an account of a licensee's periodic safety review that is to serve as the basis of the Authority's position on long-term operation needs to encompass analyses describing the facility's ageing status over time for certain key parameters, such as irradiation embrittlement of reactor pressure vessels, fatigue of reactor components and loss of prestress in the reactor containment tendons. This also applies to analyses and when checking the status (condition-based monitoring) of safety-critical electrical cables, instrumentation and control equipment.

SSM's regulations already impose requirements on licensees' ageing management, but they need to be clarified and worded more precisely in various respects. SSM also needs to review its regulatory supervision so it is better adapted to the matters and aspects that become relevant in connection with long-term operation. Proposed changes and improvements to regulations and supervision when it comes to ageing and ageing management issues are described by report [3]. See also section 4.5.4 below.

4. An analysis of the Swedish regulatory model in the field of reactor safety

4.1. Starting points

The stipulated aim of the Act on Nuclear Activities when it comes to safety work is to as far as possible eliminate the risk of a radiological accident and thus, ultimately, the loss of life and property. This is why the Act on Nuclear Activities has been formulated to give the licensee a virtually strict liability for conducting nuclear activities. This fundamental liability cannot be assigned to any other party. While taking into consideration the potentially serious impact on life, health and the environment as a result of a reactor accident, the State must nevertheless regulate and supervise the operation to ensure that safety is being maintained and that the licensee is managing the safety work in the manner intended by the Act on Nuclear Activities. Both regulation and regulatory supervision need to encompass not only technical aspects at the facilities, but also aspects of human resources, organisations and administration related to safety work.

SSM currently has a satisfactory model for regulation and regulatory supervision in the field of reactor safety, but which needs to be developed in different respects. This model mainly represents regulation and regulatory supervision focusing on licensees' management and control of their activities and which has evolved over the past 20 years, primarily within the regulatory authority of the time, the Swedish Nuclear Power Inspectorate (SKI). The principles for this model were established in a situation where all Swedish nuclear power plants were to have been shut down by 2010. Among other things, this situation implied that the focus of the regulatory model came to be placed on safety issues related to operation and maintenance. Part of this model includes a general kind of regulation imposing generally worded requirements on the work activities at Swedish nuclear power plants, also with supervisory work oriented at the licensees' management, control and follow-ups of the organisation's work.

The situation and assumptions for this focus subsequently changed by means of the Riksdag's position in 1997 owing to the Government Bill 'A Sustainable Energy Supply' that for example resulted in the Riksdag supporting the Government's proposal concerning the closure of the two nuclear power reactors at Barsebäck and not setting a specific year for taking the last nuclear power reactor in Sweden out of operation. Wide-ranging efforts were launched on safety upgrades of the ten remaining nuclear power plants and further power uprates of their reactors. The modernisation work resumed in earnest in 2005 because new regulations entered into force. Also, the licensees began planning for ‘long-term operation’, that is, operation exceeding the 40 years or so that the facilities were originally analysed and designed for. This created the need to change the regulatory framework and regulatory supervision so that they encompassed more elements of technical requirements and reviews.

The greater extent of regional and international nuclear safety co-operation has also indicated a need to change and develop the Swedish regulatory model within this area. Overall harmonisation of requirements takes place through the joint projects of nuclear regulatory authorities in the European Union as part of the Western European Nuclear Regulators’ Association (WENRA), which is also a forum for areas concerning good supervisory practices. Internationally, equivalent co-operation on good supervisory practices takes place within the framework of the Committee on Nuclear Regulatory Activities, or CNRA. The International Atomic Energy Agency’s (IAEA) development of requirements and standards for the work of radiation safety authorities is increasingly contributing to a more harmonised point of view on regulation and regulatory supervision, for instance in terms of reactor safety issues. The European Union’s Nuclear Safety Directive imposes requirements on Member States to, at least every ten years, undergo a review in relation to the IAEA’s documentation regarding regulation and regulatory supervision of national nuclear safety and radiation protection. The results are to be reported both to the European Commission and to the Member States. Exchange of experience between regulatory authorities concerning these results is intended to take place within the European Nuclear Safety Regulators Group, or ENSREG.

The establishment of SSM, the Swedish Radiation Safety Authority, launched extensive revision of the existing regulatory model. The partially changed assumptions for the regulatory model applied up until that time to licensees’ management and control of their activities in the field of reactor safety, as well as increased harmonisation ambitions regionally and internationally, in addition to the results from the IRRS (Integrated Regulatory Review Service) peer review that was performed in February 2012 on activities in Sweden and on SSM’s work, have now accentuated the need for further development and changes.

In the investigative report [4], SSM provides an analysis of the Swedish regulatory model based on the IRRS results, international standards, recommendations and practices. What’s more, documents produced within the framework of the CNRA and WENRA have been taken into account. This analysis is also based on a general survey of trends in regulatory supervision observed at a number of sister authorities in other countries. This is summarised in sections 4.2 to 4.6 below.

4.2. Regulatory supervision in the field of reactor safety

4.2.1. Basic principles for supervision

When it comes to the field of reactor safety, SSM has largely continued with the regulatory model developed by SKI applying to licensees’ management and control of their activities. The aim of SSM's supervision is to assess the capability of the parties carrying out an activity to lead and manage their work from the perspective of radiation safety. This implies that parties carrying out activities must have a suitable management and governance, including self-assessments of a high standard, giving the desired impact. SSM's supervision may be comprehensive, for example by checking management systems, or be conducted on a detailed level by e.g. supervising particular practices.

The aim of supervision is to verify that radiation safety is being maintained and improved. This is done by:

- checking compliance with acts, ordinances, regulations, conditions and other requirements, and
- monitoring the parties' activities as a basis for proactive and preventive work.

4.2.2. Regulatory supervision of nuclear power plants

Today, regulatory supervision of safety at nuclear power plants is conducted in the form of surveillance activities, compliance inspections, reviews and follow-ups of events. The results from these activities provide input both for decision-making concerning various measures as well as for the integrated safety assessments, which as far as possible are to paint a composite picture of nuclear safety and radiation protection at the facilities. Compliance inspections and reviews can lead to questions being posed requiring more in-depth investigation until the Authority can adopt a standpoint on observations made or new concerns. For this reason, regulatory supervision and licensing reviews are underpinned by investigative and research work, in addition to international co-operation in the field of nuclear safety and radiation protection. The results from this work also serve as input for the integrated safety assessments. Other input is provided by research and investigative work, beyond outcomes from regulatory supervision and the integrated safety assessments, to enable continuous monitoring of the suitability of the nuclear safety and radiation protection standards applied and to keep them up to date.

<u>Surveillance activities</u> are key elements of regular supervision. Objectives of surveillance inspection activities include:

- maintaining an overview of the parts of activities of significance for safety and monitoring these activities' development and changes over time
- at an early stage, identifying planned modifications in addition to indications and deficiencies that may need more in-depth checking in the form of inspections or reviews
- following up decisions in cases where SSM has (for example) imposed a requirement on an action plan and where there is a need for random sampling to a smaller extent but not needing to be done in the form of inspections nor reviews
- providing information and communicating SSM's point of view in various areas while at an early phase clarifying SSM's requirements and defining expectations placed on the licensee

Surveillance activities can take place both announced and unannounced.

<u>A compliance inspection</u> is one of the tools of regulatory supervision used to assess the compliance of a licensee with the requirements and conditions imposed on a particular activity. One aspect of this work involves assessing the standard of safety work and one's

capability to work preventively, deal with deviations and learn from experiences, all with the aim of maintaining safety and working continuously to improve it.

A compliance inspection involves SSM systematically compiling information about the inspected work activity. This usually takes place in the form of an initial review of the parts of the facility's management and quality assurance systems governing the organisation and management of the relevant activity. This review is followed by structured interviews with facility personnel in order to survey the existing practices actually being applied.

A rapid investigation is the supervisory tool used when an event has occurred or a circumstance has been discovered requiring more information before the Authority can decide on supervisory measures. The aim of a rapid investigation is to enable the Authority to gain a quick and independent interpretation of the event.

A review is another of SSM's tools of regulatory supervision used to assess the compliance of a licensee with the requirements and conditions imposed. Reviews are performed on notified technical modifications of a facility and organisational changes, or changes to its work activities, in addition to changes to documentation that is of central importance to safety. Reviews are also performed on supporting documentation belonging to licence applications and requests for authorisation of different kinds, as well as requested exemptions from regulatory requirements.

SSM's reviews are made in the form of an analysis and assessment of the substantive issues touched upon in the matter. This also includes assessing how a facility's self-assessment has been performed in a particular case.

Experience feedback including follow-ups and assessments of safety-related events occurring at the facilities is an important part of regular supervision. SSM's regulations contain provisions governing investigation, reporting and classification of this kind of event occurring at Swedish facilities. There are also international reporting systems (called 'IRS' and 'INES') to which SSM reports events in Sweden and, in its turn, receives information from about events that have occurred in other countries and that might be of significance when assessing conditions at Swedish facilities.

SSM conducts an assessment of the licensees' own analyses of the events that have occurred at Swedish facilities, the conclusions drawn and action taken to prevent reoccurrence.

SSM's integrated safety assessments are performed to achieve a holistic perspective that separate instances of supervisory action cannot provide. This perspective is a necessity to enable SSM to express an opinion on the overall impact of a number of measures being taken by the licensees over a period of time, also whether they are improving safety in the way intended by acts, regulations and licence conditions. The holistic perspective is also needed for timely detection of deficiencies on the basis of relatively minor events and deviations that, viewed separately, are not serious, but which show a pattern when viewed as a whole. What's more, the integrated safety assessments provide key input for SSM's prioritisation of its supervisory work.

The integrated safety assessments culminate in the results from all supervisory activities pertaining to a particular licensee and its nuclear power plants being compiled annually and systematically evaluated in relation to the requirements imposed on the assumptions for and the different levels of the plants' defence in depth. Results from SSM's investigative and research work that may be of significance when evaluating the facilities and licensee activities are also incorporated into the integrated safety assessments.

As described in section 3.6 above, periodic safety reviews of the facilities are also performed every ten years. These reviews are conducted in relation to developments in science and technology. They are based on analyses and reports in which the licensee is to provide an account of:

- the way in which the facility's design, function, organisation and operations fulfil the requirements imposed by the Act on Nuclear Activities, the Environmental Code and the Radiation Protection Act in addition to regulations and conditions issued under this legislation, and
- the prerequisites for compliance with these regulations and conditions up to the next periodic safety review.

A Swedish periodic safety review corresponds to what is internationally referred to as a 'Periodic Safety Review' (PSR).

4.3. International regulatory and supervisory co-operation

4.3.1. General information about collaboration between regulators in the field

There is extensive co-operation between radiation safety authorities in the area of regulation and supervisory work in the field of reactor safety. The regulatory authorities take part in the International Atomic Energy Agency's (IAEA) work on safety standards and standards for regulation and supervision. Internationally, radiation safety authorities co-operate within the framework of (for example) the OECD's nuclear energy body, the Committee on Nuclear Regulatory Activities, or CNRA. In the EU, regulatory authorities co-operate in the Western European Nuclear Regulators' Association (WENRA) and the European Nuclear Safety Regulators Group (ENSREG).

4.3.2. The IAEA's requirements and recommendations

The IAEA has drawn up a number of documents containing 'requirements' and recommendations for the organisation, control and regulatory supervision of nuclear safety and radiation protection in a country. The IAEA has five main documents pertaining to legal matters and regulatory authority aspects in the nuclear field.

The set of safety requirements, "Governmental, Legal and Regulatory Framework for Safety: General Safety Requirements, Part 1", is, among other things, used during IRRS reviews of countries and regulatory authorities performed by the IAEA which, under the Nuclear Safety Directive (2009/71/EURATOM), are required for all Member States a minimum of every ten years as of 2014. See also below in the section about IRRS reviews.

As indicated by its title, this document covers fundamental requirements imposed on the legal framework and functions of regulatory authorities in the area of nuclear safety and radiation protection. The requirements are essential for fulfilling the IAEA's Safety Fundamentals.

Thus, this document contains fundamental requirements on legislation in the area and stipulates that there must be one or more supervisory authorities in the area. There are also fundamental requirements on their role, tasks, responsibilities and mandate, also their independence from other organisations with a responsibility for nuclear activities or work activities relating to radiation. In cases where there are several regulatory authorities in the area, another requirement is clearly defined conditions for responsibilities and co-operation, for example to avoid overlapping activities in relation to, or contradictory requirements on, the relevant licensees and related organisations.

The fundamental tasks of the regulatory authority, or authorities, are to be defined by national legislation, and shall, depending on the nuclear activities and the activities involving radiation in the country, encompass licensing reviews and granting of licences, safety reviews, safety evaluations, compliance inspections and issuing regulations and sanctions. This is to apply throughout all stages, from applying for authorisation to construct and operate a facility, to its decommissioning and management of radioactive waste. The mandates of the regulatory authority are also specified in terms of sanctions and access to information.

Fundamental requirements are also imposed on the authority's organisation and staffing as well as other resources. This is related to the overall scope of the operations, in other words the kinds and number of nuclear activities and other regulated activities involving radiation included in the mandate for regulatory supervision. There must be a sufficient number of in-house staff of the authority having the necessary qualifications, experience and expertise in the relevant specialist areas to enable both integrated periodic safety reviews as well as specialist assessments in different areas. If the authority lacks in-house expertise in all the relevant specialist areas, then the authority is allowed to turn to consultants for assistance. These consultants must be independent of the licensees and related organisations. If there is a lack of consultants nationally who are sufficiently independent, assistance should instead be sought from sister authorities abroad or independent and internationally recognised expert organisations. However, the authority must not be dependent on the consultants' assessments, nor obviously on the assessments of the licensees or parties carrying out activities. Such assessments must always be subject to an evaluation by the authority's own personnel. No matter the circumstances, the authority's responsibility is permanent, meaning that decision-making and adopting a position on safety issues must not be delegated.

The requirements stipulate that the work of an authority on licensing reviews and granting of licences, safety reviews, safety evaluations, compliance inspections and issuing regulations is to be governed by the processes of a management system. Comprehensive requirements on the content of the different parts of a management system are also stipulated. The IAEA's set of safety requirements also lays down well-defined and clear responsibilities within the authority and that the authority must take part in international co-operation and exchange of experience in the area.

The authority's safety reviews, safety assessments and compliance inspections must be performed systematically while also being well managed and planned. The document points out that reviews are to be conducted for the purpose of verifying and checking that activities and facilities fulfil the requirements imposed. It is also defined on a comprehensive level the scope and orientation of the authority's review and evaluation work. Key aspects to take into consideration during reviews are also described. This area is further expanded upon in the standard “Review and Assessment of Nuclear Facilities by the Regulatory Body”.

In a similar way, the IAEA requirements point out that inspections are to be conducted for the purpose of verifying that activities and facilities fulfil the requirements imposed. Both pre-planned and reactive inspections are to be conducted in addition to inspections owing to events that have occurred. Such inspections can be either announced or unannounced. The main kinds of activities to be inspected are stated on an overall level. The IAEA’s point of view on inspection work is further expanded upon in the standard “Regulatory Inspection of Nuclear Facilities and Enforcement by the Regulatory Body”.

The regulatory system is to define how the authority takes decisions on sanctions as a means for achieving rectification. Sanction measures are to be set in proportion to the safety significance of deviations from the requirements (non-compliance) and should encompass everything from warnings to lost licences.

4.3.3. The CNRA’s projects on development of regulatory authorities

The nuclear regulatory authorities belonging to the OECD have for a long period of time now co-operated through the CNRA in areas relating to both their prescriptive work and supervisory work. Other areas discussed and dealt with include the orientation, effectiveness and efficiency of these activities. Investigations and comparative studies have also been conducted. The results of these efforts have been compiled as part of the report entitled “Improving Nuclear Regulation”.

In these contexts, the CNRA has made it clear how to view the authorities’ effectiveness, efficiency and capability, in addition to the impact of the work by means of the following distinctions:

- Regulatory *effectiveness* means “to do the right work”
- Regulatory *efficiency* means “to do the work right”

In the opinion of the CNRA, a regulatory body is effective and efficient, having been given the necessary authority and resources for its tasks, when it:

- through its operations, ensures that an acceptable level of safety is being maintained by the regulated operating organisations and takes appropriate actions when there are signs of degradation of safety
- promotes safety improvements among regulated operating organisations
- performs its regulatory functions in a timely and cost-effective manner
- performs its regulatory functions in a manner that ensures the confidence of the general public and mandator (government/parliament)
- develops and maintains an adequate level of competence for its tasks
- strives for continuous improvements in its performance

The supervisory areas studied by the CNRA include:

- assessments and investigations of safety culture
- regulatory use of operating experience
- regulatory control over licensees’ self-evaluations/self-assessments
- regulatory control over licensees’ suppliers and contractors
- assessments of safety upgrades

The CNRA has given recommendations in these areas about the scope and orientation of regulatory supervision. Key aspects of regulatory supervision are discussed, and in some instances, recommendations are also provided concerning relevant phases when regulatory bodies should intervene, and the method for this. When it comes to regulatory use of operating experience, recommendations are provided not only about regulatory supervision of a licensee's system for event management, but also the regulatory bodies' own work on event analyses and drawing conclusions of importance for regulatory supervision and the regulatory framework.

In this document, the CNRA stresses the importance of the licensees' management and follow-ups of their suppliers and contractors, also that regulatory supervision should for this reason have a particular focus on licensee systems and capability for such management and follow-ups. This especially applies to the current situation, with suppliers and component manufacturers that are increasingly global. The CNRA report also highlights the need for regulatory supervision to encompass follow-ups to look into the performance of licensees' management and quality assurance work on site at manufacturers and suppliers. This area is also highlighted in a recently published report [21] from the CNRA Working Group on the Regulation of New Reactors (WGRNR) as well as in the IAEA's IRRS review of SSM's work activities; see also section 4.4 below.

In June 2010, the CNRA set up its Senior Task Group with the aim of developing recommendations for the regulatory authorities' work on adopting a position on, and supervision of, what is referred to as 'long-term operation'. As explained in Chapter 3, long-term operation (LTO) refers to operation beyond the period of time for which the facilities were originally designed and analysed. The reason why the CNRA launched this work is because long-term operation of nuclear power plants is being planned or has begun in many countries to meet their demand for electrical power and to meet the climate targets for emissions of greenhouse gases. In some countries, long-term operation is viewed as a phase to fill the gap until today's reactors have been replaced by new ones, or until the emergence of some other new kind of production capacity. Long-term operation presents both licensees and regulatory authorities with new challenges, starting with the requirements imposed on safe operation during the extended period of operation.

In its final report, "Challenges in Long-term Operation of Nuclear Power Plants - Implications for Regulatory Bodies", the task group pointed out that:

- the organisation, resources and expertise of the regulatory bodies must be adaptable to manage new safety issues that might arise in connection with long-term operation
- regulatory bodies' regulations need to be revised so that they place a clearer focus on aspects of ageing and the licensees' ageing management activities, as well as gradual upgrades to increase the safety margins against unforeseen events at ageing facilities

The task group also pointed out the need for clear and formal processes, both for adopting a position on long-term operation as well as for regulatory supervision over the extended period of operation. For countries such as Sweden that do not limit operating permits to a specific period of time, the recommendation is to adopt a position here within the framework of a periodic safety review (PSR).

This also means that regulatory bodies, once long-term operation has been decided, will also need to focus on ageing issues in many different respects. This applies to physical ageing of systems, structures, components and equipment as well as technological ageing. Regulatory supervision also needs to place a focus on management and control of the work activity in addition to competence issues in the area.

4.3.4. Recommendations from the WENRA Inspection Group

One of the main tasks of WENRA is to achieve consensus in different areas of the nuclear safety field. This work is carried out in the form of ‘reference levels’ and good practice.

In 2010, WENRA set up a working group, the WENRA Inspection Group, for the purpose of surveying the practices currently applied by the regulatory bodies for reviews and inspections in connection with design, manufacturing, installation and commissioning of pressure-bearing and other mechanical components in addition to building structures. The work of the Group also included using this analysis as a platform for discussing and developing good practice that could serve as the basis for a common approach to the extent that this is possible while taking into account legal points of departure and fundamental philosophies of regulatory supervision.

This recommended approach is described in the WENRA report “Benchmarking the European inspection practices for components and structures of nuclear facilities” and is based on the following circumstances:

- Issuing regulations and regulatory supervision in the area of design, manufacturing, installation and commissioning of pressure-bearing and other mechanical components in addition to building structures are often prescriptive/normative (see also section 4.5 below) in many countries.
- Many regulatory bodies in this area prescribe that certain reviews must be performed by independent inspection bodies. These bodies are also to monitor certain steps of work during manufacturing, installation and commissioning.

‘Independent inspection bodies’ refers to organisations that are independent of manufacturers, suppliers and licensees. The organisations also need to have sufficient resources and demonstrated professional skills in the area while also being accredited and approved for the tasks in question. Accreditation is granted by an accreditation body. Beyond this, it might also be relevant for the national radiation safety authority to review and assess the inspection bodies’ independence, resources and professional skills in the area.

By using this kind of accredited/approved independent inspection body, a system of control can be applied where the bodies carry out many review and monitoring tasks in connection with design, manufacturing, installation and commissioning of pressure-bearing and other mechanical components in addition to building structures. As a consequence, the radiation safety authorities can then limit their reviews and supervisions to principal areas, design assumptions and design bases in addition to assignments in connection with components, systems and facilities beginning test operation and being commissioned.

On the other hand, additional work for the radiation safety authorities involves taking part in accreditations, and when applicable, own approval procedures as well as annual audits of the inspection bodies in order to check that the bodies are performing their tasks to the extent and in the manner necessary.

This approach is closely linked to the system applying in many sectors of society that are subject to Community legislation of the European Union in the form of product directives and harmonised standards, though with the difference that the field of reactor safety is subject to national frameworks, and there are so far only harmonised standards to a very limited extent.

This is also an approach whose main elements have been applicable over a long period of time now in parts of the nuclear field in Sweden.

4.4. IRRS peer review of Sweden

Between 6 and 17 February 2012, an IRRS review was performed of activities in Sweden and at SSM in the fields of nuclear safety and radiation protection. The review mission was mainly a 'full scope review', that is, the review mission largely encompassed the entire fields of nuclear safety and radiation protection. However, the areas of physical protection and non-ionising radiation were not reviewed.

Prior to the review mission, SSM conducted a self-assessment of its work activities. Part of the self-assessment included responding to a very large number of questions compiled by the IAEA on the basis of the set of safety requirements and standards mentioned in section 4.3.2, and conclusions were drawn on how the Swedish system and regulatory authority fulfil the requirements and meet the standards. This self-assessment was one of several forms of input used by the IRRS peer review team.

The IRRS peer review was conducted by 18 experts from 16 countries in addition to six experts from the IAEA. The peer review team's overarching conclusion was that the Swedish system for nuclear safety and radiation protection is stable and well-developed, with e.g. an independent supervisory authority that is open and transparent, that learns from experience and is open to feedback. The team emphasised good practice as well as areas of improvement. The team presented 15 examples of 'good practice' that may be relevant for other countries and regulatory authorities to learn from. In the field of nuclear safety, this was for instance about SSM having its own good MTO (Man-Technology-Organisation) expertise that is used as part of regulatory supervision and that SSM's regulation has brought about a wide-ranging safety upgrade programme. The team also submitted 22 'recommendations' and 17 'suggestions' on improvements to the Swedish system.

Eight of these recommendations were directed at the Swedish Government, urging it to, among other things:

- take measures to maintain national competence in the fields of nuclear safety and radiation protection
- increase SSM's resources for regulatory supervision and licensing reviews
- establish an ongoing process that keeps legislation up to date
- ensure that SSM has legal potential to conduct inspections of suppliers
- clarify the mandate and authority for the purpose of withdrawing/terminating licences

Important recommendations and suggestions that directly or indirectly relate to SSM's work vis-à-vis nuclear power plants concern the regulations, SSM's management system with the internal management process and guidance, in addition to resources and securing competence.

As far as concerns SSM's regulatory framework for reactor safety, the team established in the report that Swedish regulations concerning nuclear facilities have evolved historically in pace with the need for regulation. The peer review team has also established that the IAEA's safety standards were being used as a basis for Swedish nuclear safety regulations, or the regulations have referred to them, but not in a systematic way. The peer review team highlighted examples of this by pointing out areas/sub-areas considered to be unsatisfactorily regulated in relation to what ensues from the IAEA's safety standards. These examples cover regulations for reactor containments, electrical and control equipment, taking into account external events, probabilistic safety analysis (PSA), safety classification, fire safety, management systems and designing nuclear power plants. For this reason, the team recommends that SSM revise the present regulations to make them clearer, more consistent and comprehensive.

As far as concerns SSM's regulatory supervision, the review team established that the inspection activities largely focus on licensee management systems and their use in practice by interviewing personnel at different levels of the facilities' organisations and in different functions. The team also established that SSM performs few unannounced compliance inspections at the nuclear power plants. This is why SSM is recommended to consider performing more unannounced compliance inspections and inspections in more areas while also observing activities and performing technical inspections more frequently.

In the IAEA's General Safety Requirements, GSR Part 1, it is stated that national regulatory bodies are to analyse and evaluate events and other safety-related operating experience, nationally and internationally, so that they can draw conclusions that may be of importance for regulatory supervision and the regulatory framework. The peer review team established that in terms of nuclear power plants, SSM has a satisfactory system for following up events that take place at Swedish facilities. However, the team noted that the Authority does not disseminate information to the relevant stakeholders about events that have occurred, nor is there a formalised system for informing the parties concerned about the conclusions drawn by SSM from events that have occurred. This applies to cases where the events have not resulted in a particular inspection nor review. Here, SSM disseminates the reports to the relevant licensees.

It emerged from both SSM's self-assessment and the IRRS review that the Authority's management system needs to be developed further in different respects. SSM needs to develop more specific guidance on how to pursue work on various review and inspection tasks. These processes and written routines should also be communicated to applicants, licensees and other interested parties. The peer review team has also recommended that SSM consider introducing formal competence requirements and compulsory training programmes for all personnel with supervisory tasks, especially for different kinds of technical experts.

Based on the results from the self-assessment and IRRS review, SSM has produced an action plan showing how and when the Authority is to rectify deficiencies and tackle areas of improvement.

4.5. An international exploratory study of regulatory strategies in the field of reactor safety

4.5.1. Identification of shared patterns, approaches and strategies

Despite the fact that regulatory supervision is conducted in many different sectors of society both in Sweden and internationally, little research has been done into different strategies and methods of this kind of regulatory control. This situation was also pointed out by the inquiry into regulatory supervision in its final report containing proposals for clearer and more effective regulatory control (*Tillsyn. Förslag om en tydligare och effektivare tillsyn,* SOU 2004:100). This led to SKI launching research activities in this area in the early 2000s. One of the projects launched had a specific focus on regulatory approaches and strategies in the field of reactor safety. In this context, 'regulatory approaches and strategies' refers both to strategies when it comes to regulations and related advice/guides, and to regulatory supervision, i.e. oversight.

The findings from this research are described in the report entitled "Experience with Regulatory Strategies in Nuclear Power Oversight: An International Exploratory Study". As can be seen from the title, the purpose of this exploratory study was to identify approaches and strategies applied by supervisory authorities in the field of reactor safety. Contributors included experts from regulatory bodies[12] in Canada, Finland, Spain, Sweden, the United Kingdom and the United States. The intention of the study was not to make a comparison between the authorities' regulatory strategies.

The study initially identified six main regulatory approaches applied by these authorities. Shared patterns were also identified in terms of:

- the experts' point of view on the main advantages and difficulties associated with use of the respective strategy
- the experts' experiences from applying the strategies as part of regulation and regulatory supervision of plant modifications, management systems in addition to training and skills assessment of operations personnel
- the experts' perceptions on the effects and consequences of the different strategies

The findings from these investigations were then discussed during a workshop with representatives from the attending regulatory authorities. During this workshop, additional shared patterns and approaches were identified. This resulted in the following regulatory approaches being defined, which were to varying extents applied by the authorities:

- A prescriptive orientation in terms of regulation and regulatory supervision
- Facility-based regulation and regulatory supervision
- Outcome-based regulation and regulatory supervision
- Risk informed regulation and regulatory supervision, or regulation and regulatory supervision oriented at hazard potential (hazard informed)
- Process-based regulation and regulatory supervision
- Self-assessment based regulation and regulatory supervision
- Influence or education based work by the regulator

<u>A prescriptive approach</u> by the regulator means that it establishes specific requirements on technical solutions as well as on how activities are to be conducted, plus its checking compliance on a more detailed level.

The main benefit of this approach is that it is spelled out clearly to the licensee about the requirements imposed and the regulator's expectations. The main difficulties are that this

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[12] Canadian Nuclear Safety Commission (CNSC), Consejo de Seguridad Nuclear (CSN), Nuclear Installation Inspectorate (NII), now the Office for Nuclear Regulation (ONR), the Finnish Radiation and Nuclear Safety Authority (STUK), the Swedish Nuclear Power Inspectorate (SKI) and U.S. Nuclear Regulatory Commission (NRC).

approach may tend to take responsibility away from the licensee, it requires a high use of regulator resources, and the system can become rigid and difficult to change.

Facility-based regulation and regulatory supervision implies that the regulator imposes no generally applicable requirements; instead, it imposes specific requirements on each facility based on its particular design, layout and activities. The focus of regulatory supervision is then on follow-ups of compliance with the specific requirements.

The main benefit of this approach is that it allows greater flexibility because the requirements are adapted from case to case. The main difficulties are that the approach can be perceived as arbitrary and become inconsistent when imposing requirements on similar facilities. What's more, this strategy demands a relatively high level of resources from the regulator for the specific adaptations of requirements and subsequent follow-ups as part of regulatory supervision.

Outcome-based approach to regulation and regulatory supervision implies the authority establishing specific performance goals, or outcomes, when it comes to facility safety and the operation, but without specifying how they are to be obtained. Licensees determine how they will achieve these outcomes.

The main benefit of this approach is that it allows the licensee to determine the best way of conducting the activity in order to achieve the safety goals and safety results. The main difficulties for regulators are to identify and specify the goals and outcomes needed to maintain safety, and then as part of regulatory supervision, identify ways to conduct follow-ups and measure against the goals.

Risk-informed regulation and regulatory supervision and hazard-informed regulation and regulatory supervision imply the regulator identifying areas of greatest risk for different parts of a facility and its activities, and these areas receiving priority for regulatory attention. Risk-informed and hazard-informed parts are assigned priority when specifying requirements and as part of regulatory supervision.

The main benefit of these approaches is that the requirements imposed can be set in relation to the level of risk and hazard while also orienting regulatory supervision thereafter. In this way, regulatory supervision can be made optimal. The main difficulties of these approaches are to determine the reliability of the methods of risk evaluation/risk analysis and the methods for evaluating hazards as well as assessing the input data that may or should be applied. According to the attending experts from regulators, these approaches should consequently only be used in combination with other approaches.

A process approach implies the regulator identifying the key processes that the licensee should apply in order to maintain safety at the facility and lead to safe performance, and subsequently requiring the licensees to establish and implement these processes. Regulatory oversight focuses on following up the licensees' compliance and the processes' suitability and effectiveness.

The main benefit of this approach is that it gives the regulator a more in-depth understanding of a licensee's activities and the factors affecting its safety work. The main difficulty of this approach on the part of regulators is defining the safety goals at various levels of a management system in relation to which the processes and their effectiveness are to be evaluated. This is a complex task.

Self-assessment approach to regulation and regulatory supervision implies the regulator requiring the licensee to develop and implement a system for continuous own follow-ups, reviews and evaluations of its safety work. The regulator then focuses on the licensee's systems and carries out targeted measures to assess how the systems are applied and their subsequent outcomes.

The main benefit of this approach is its inherent implication that the primary responsibility for safety is put on the licensee. The main difficulty of this approach is that it can raise issues of credibility with the general public when it comes to the regulator's work, so it should be used in combination with other approaches. What's more, the regulator must conduct a suitable level of own in-depth assessments of results from a licensee's self-assessment work. This can be resource-intensive.

Influence or education approach of the regulator implies the authority arranging training activities, workshops, seminars and other forms of informative action with the aim of influencing a licensee's safety work in different respects. According to the attending experts, this approach should only be used in combination with other strategies in terms of regulation and regulatory supervision.

4.5.2. Key conclusions from the first study

This project involved studying and discussing application of these approaches in the following three 'functional areas of oversight': regulatory oversight of plant modifications, regulatory oversight of the licensees' management systems, regulatory oversight of training, and qualification of reactor operators/working staff. Here are some key conclusions:

- None of the approaches were used on their own. Depending on the area and supervision matters, combinations of at least two, and often three and sometimes four different approaches are used in order to achieve an adequate regulatory strategy shaped to fit the nature of the area.
- Use of the prescriptive approach is generally declining in prevalence while the process-oriented approach is increasing. When safety problems arise and when regulators are of the view that measures need to be taken, it is common for the prescriptive approach to be used.
- The risk-informed and hazard approaches can be used in many areas of regulatory oversight so as to enable prioritisation and optimisation of regulators' work.
- It was difficult to achieve a common understanding among the attending experts on the extent to which a risk informed approach should govern regulatory oversight, particularly in areas concerning facility design and plant modifications. A balanced combination between prescriptive and risk-informed approaches is preferable in these contexts.

4.5.3. Follow-up study

Major changes have taken place in the nuclear power industry internationally since the above-mentioned study was conducted. Long-term operation (LTO) is being planned for in many countries, which has an impact on the orientation of regulation and regulatory supervision. Against this background, issues relating to ageing management as well as issues concerning safety modernisation/safety upgrades need to be taken into account. New nuclear power plants are being built in many countries and the licensees are continuing to optimise operations and maintenance. Generational shifts are taking place at both licensees and regulators. The ways of leading and managing activities at nuclear power plants are changing. Altogether, this is leading to significant, and to some extent, new challenges for the supervisory authorities. With this in mind, SSM launched a follow-up study in 2011 involving experts from the same regulatory authorities and countries to (for example) provide additional input for an analysis of the Swedish regulatory model and in order to develop SSM's work. The follow-up study focuses on the following eight regulatory approaches:

1. Regulatory supervision of management systems at operating nuclear power plants
2. Regulatory supervision of operations at operating nuclear power plants
3. Regulatory supervision of maintenance activities at operating nuclear power plants
4. Regulatory supervision of safety culture at operating nuclear power plants
5. Regulatory inspection and assessment/supervision of major plant modifications
6. Regulatory inspection and assessment/supervision of power uprates at nuclear power plants
7. Regulatory inspection and assessment/supervision in connection with decisions on long-term operation (LTO) of nuclear power plants
8. Regulatory supervision of construction and commissioning of new nuclear power plants (excluding the regulatory activities for issuing a licence)

This study is based on a questionnaire filled out by 54 experts in the area from nuclear power regulators in five countries: Canada, the United Kingdom, Sweden, Spain and the United States. These experts responded to questions belonging to their field and provided their experiences with regulatory approaches and the outcomes from the approaches used in the area of regulatory supervision. The study will also encompass follow-up interviews. The study is based on the same regulatory approaches as applied by the previous study. It has been estimated that this work can be finalised in the spring of 2013[13] in workshop form.

Altogether for the eight areas studied, the most common regulatory approach is the self-assessment approach. Just under half of the experts responded that other common approaches were the outcome-based, risk- or hazard-based, and the process-based approaches. The least common one is the facility-based approach.

Divided by functional area of supervision, the picture appears to be somewhat different, with the results largely as shown in the figure below illustrating the approaches most often used and commonly used for each individual area.

Approach Functional area	Prescriptive	Facility-based	Outcome	Risk/hazard	Process	Self-assessment	Influence-based
Management system							
Operation							
Maintenance							
Safety culture							

[13] Postponed until autumn 2013.

Major plant modifications							
Power uprates							
Long-term operation (LTO)							
Construction of new facilities[1)] ('construction phase')							

1) Only three experts responded to the questions in this area, making the outcome inconclusive.

Approaches most often used

Approaches commonly used

Thus, this illustration also confirms the findings from the first study showing that approaches and strategies having combinations of approaches depend on the nature of regulatory supervision. There is also a tendency to combine additional approaches as a functional area of regulatory supervision becomes more complex. Moreover, it has been shown that a strategy having a combination of the prescriptive, process-based and self-assessment approaches is used in nearly all areas, but having somewhat differing areas of emphasis.

4.5.4. Clearer and more comprehensive regulation

During the early phases of the Swedish nuclear power programme, there were no generally applicable regulations. Licence conditions worded as regulations had been issued in certain areas, such as the licensees' quality assurance work and for pressure-bearing and other mechanical components.

It was first through the 1992 amendment of the Act on Nuclear Activities (1984:3) that the former regulator, the Swedish Nuclear Power Inspectorate (SKI), was given the general powers to issue regulations and a regulatory code. A commission of inquiry for an international review performed between 1995 and 1996 of SKI and SSI's operations recommended in its final report, "Swedish Nuclear Regulatory Activities", Swedish Government Official Report SOU 1996:73, that SKI should prioritise work on developing its regulatory framework. The amended Act and these recommendations led to SKI producing a set of regulations with associated general advice. These regulations, which were later incorporated as part of SSM's Regulatory Code in conjunction with the merger between SKI and SSI, are expressed relatively generally, and in some cases, are targeted specifically. This was a deliberate choice that took into account the regulatory model applied to licensees' management and control of their activities. This model was introduced by the authority following recommendations from the international review of 1995–1996.

Experiences from applying this model have, however, demonstrated the ongoing need to clarify the requirements, while also conducting revisions and making additions in certain areas. As mentioned in section 4.4, the IRRS review also established that the Swedish regulatory framework for nuclear facilities has evolved in pace with the need for regulation. The peer review team also established that the IAEA's safety standards were being used as a basis for Swedish nuclear safety regulations, or were referred to in the regulations, but not in a systematic way. For this reason, the IRRS peer review team recommended that SSM review the present regulations and revise them to make them clearer, more consistent and comprehensive.

The comparison made between the regulations and general advice of SSMFS 2008:17 concerning the design and construction of nuclear power reactors in relation to corresponding requirements in Finland, the United Kingdom, Canada and the United States (as described in section 2.6.1.) also demonstrates the need to provide more precise wording and clarifications in the Swedish regulatory framework. As explained in section 2.8 and Chapter 3, there is also a need to clarify the legal requirements in various respects when it comes to measures to ensure resilience against external events and resilience against ageing phenomena during long-term operation.

Thus, these experiences and results clearly demonstrate a need to revise the regulatory framework in its entirety in the field of reactor safety. This is a project that is part of SSM's action plan as a result of the IRRS recommendations and that will be highly prioritised over the coming years. The objective of revising the regulations is to make them more comprehensive and to achieve better structure, clarity and predictability about the requirements that apply. The regulations will also be formulated so that they effectively underpin the Authority's supervisory work.

The challenge for SSM will be to supplement and clarify regulations and general advice concerning nuclear power safety to enable achievement of these objectives while not going too far in terms of detailed regulation in a way that leads to unclear circumstances about a licensee's safety responsibility. This means that a future regulatory framework should have a focus on requirements oriented toward functions and characteristics, while also providing clear guidance in the form of general advice. Key points of departure for the regulatory framework's revision include the IAEA's safety standards and the 'safety reference levels' and other documents agreed between nuclear regulatory authorities in the European Union within the WENRA collaboration.

4.6. Conclusions

SSM currently has a satisfactory model for regulation and regulatory supervision in the field of reactor safety. This model stands up relatively well in relation to international standards and practice in the field, but it needs to be developed in various respects. The present analysis shows that further development and modification need to encompass not only the regulatory framework, in the form of regulations and general advice, but also regulatory supervision. The regulatory framework needs to become more comprehensive and be based on international safety standards and European practice, while also demonstrating improved predictability about the implications of the requirements imposed. As up until now, and in agreement with the proposal of the inquiry on regulatory supervision and the Government's communication to the Riksdag (parliament), regulatory supervision is to have its foundation on the controlling element in the supervision, but with an orientation and execution that are more adapted to the varying nature of the matters of supervisory work. The objective for the further development and this change is each situation having a correct orientation of regulatory supervision that is pursued in a way that is fit for purpose and effective.

SSM will consequently, at the same time as the revision of the regulatory framework takes place, be refining strategies and approaches for conducting regulatory supervision in the field of reactor safety. In order to achieve a regulatory supervision that is more effective and even more appropriate for purpose, approaches and strategies need to be developed for various areas of regulatory supervision and be adapted to the nature of the areas and matters of regulatory supervision as well as their importance for safety. The starting point for this work will be the findings of the present studies within international co-operation concerning regulatory approaches and strategies, as described in section 4.5. This means that SSM will be defining the areas of supervision for which different supervisory strategies are to be developed. For each area, it will be defined which combinations of approaches are to be applied and serve as the basis for the strategy per area. This strategy is also to be designed so as to facilitate application of the graded approach.[14] In certain areas, consideration will also be given to recommendations from the IRRS review mission, as well as recommendations from public sector collaboration as part of CNRA, WENRA and ENSREG.

This for example applies to periodic safety reviews of the facilities that are to be conducted at least every ten years. The requirements imposed on these kinds of safety reviews have for a long time now been stipulated in the authority's regulations, but were incorporated in the Act on Nuclear Activities in 2010 on the justification that these reviews are of principal importance. Even internationally, these kinds of periodic safety reviews (PSR) are of key significance for ensuring that operational experience is taken into consideration and that new safety standards lead to safety upgrades at the facilities. In the light of planned long-term operation of nuclear power plants, SSM is presently developing its process governing periodic safety reviews. SSM also intends to give the periodic safety reviews a formal role when it comes to adopting a position on long-term operation in accordance with recommendations from CNRA and other bodies as well as analyses conducted under this Government assignment (see section 3.6).

Against the background of the Fukushima Dai-ichi accident, it may also be anticipated that PSRs will be attributed greater importance internationally in the regulatory authorities' work, e.g. for regularly evaluating the validity of the applied assumptions and design bases, as well as ensuring that new knowledge and new safety standards are applied at the nuclear power plants. As mentioned in section 2.8.4, a recommendation has been made by the independent peer review performed on the EU stress tests under ENSREG to emphasise the importance of the periodic safety reviews (PSR) of the facilities. In particular, it is emphasised that natural phenomena and other critical functions are to be evaluated as often as deemed appropriate, but at least every ten years.

Other areas that have been emphasised by the IAEA, as part of international regulatory collaboration as well as by the present report, and which will be encompassed by SSM's revision of its regulatory model, include the following:

- supervision of ageing management during long-term operation
- supervision of the licensees' control and quality assurance of work performed by suppliers and contractors
- analyses and follow-ups of operating experience
- supervision in connection with design, manufacturing, installation and commissioning of pressure-bearing and other mechanical components in addition to steel and building structures by using independent and accredited bodies

[14] In this context, 'graded approach' refers to the regulation, as well as the scope and orientation of regulatory supervision in an area, being based on the matters' safety significance. This may be expressed in terms of risk or potential consequences of deficiencies in some respect in the short or long term.

5. Overall assessment of long-term safety and regulatory supervision in the Swedish nuclear power industry

5.1. The need for further safety improvements

The Act on Nuclear Activities imposes requirements on maintaining safety by taking the measures necessary in order to prevent faults in equipment, incorrectly functioning equipment, incorrect action, sabotage or some other impact that could lead to a radiological accident. Thus, this requirement implies that a licensee is under an obligation to work continually on safety and to take measures in pace with operational experience and as new knowledge becomes available. This is worded more precisely in the Authority's regulations.

Ever since the Swedish nuclear power plants were commissioned between 1972 and 1985, safety improvements at the facilities have been made when problems have arisen and events have occurred. After the position taken by the Riksdag in 1997 owing to the Government Bill 'A Sustainable Energy Supply', which for instance led to removal of the year when the last nuclear power reactor in Sweden was to be shut down, the need for more extensive safety upgrades was accentuated as far as concerns operation over an extended period of time into the future. For this reason, the former regulator, SKI, drew up regulations concerning the design and construction of nuclear power reactors implying that this kind of safety modernisation work at the Swedish nuclear power plants was launched. The regulations (the former SKIFS 2004:2, now SSMFS 2008:17) entered into force on 1 January 2005 with certain transitional provisions. The purpose of these provisions was to give the licensees time to plan and safely perform modernisation work.

Using these regulations and decided transitional plans as a point of departure, the licensees have since then worked on analyses and measures for fulfilment of the requirements. The transitional action plans originally covered the period 2005 to 2013. This work proved to be much more complicated and time consuming than was foreseen when the licensees produced their proposals and the action plans were decided. Up until 30 June 2012, altogether for the ten reactors' modernisation programmes, approximately 60 per cent of the decided measures had been implemented. There are major differences between the reactors' progress, where for a few of the plants, a great deal of work remains to be done. It is nevertheless important to point out that the measures are of varying safety importance and scope, which is why only comparing the number of measures taken and remaining does not provide an accurate picture of the overall safety improvement nor progress of the respective action plan. It will not be possible to conduct an overall assessment of compliance on the part of each facility until all the measures have been taken and they have been reviewed by SSM. For this reason, the Authority has intensified its regulatory supervision and follow-ups of these companies' work so that remaining measures for fulfilment of the requirements imposed by SSMFS 2008:17 do not take longer than necessary for their safe implementation.

In several cases, the Authority has also drawn the conclusion that the licensees' safety analysis reports on how requirements are applied and verified are all too generally worded to give a clear and complete understanding of compliance with the regulations. Reviews have also shown that the licensees' interpretation of several of the requirements does not agree with SSM's point of view on the regulations' implication.

SSM assesses that the measures taken and planned to fulfil the requirements imposed by SSMFS 2008:17 strengthen the protective system of the nuclear power reactors' barriers, mainly through increased redundancy and separation, which is the primary purpose of the regulations. Also, when fully implemented, the measures imply a strengthening of defence in depth on the part of all plants. Another safety-related consequence of these measures, other than the purely physical modifications of the facilities, is the improved level of knowledge about the plants' characteristics that the analyses vis-à-vis the legal requirements of SSMFS 2008:17 have implied among the licensees, as well as the fact that the technical documentation concerning the plants has been improved. These conditions are crucial prerequisites for ensuring safe nuclear power plants.

SSM has, however, in the presently completed analysis of safety improvements drawn the conclusion that further measures will be needed beyond the scope of the licensees' action plans for fulfilment of the requirements of SSMFS 2008:17. The results of the updated comprehensive risk and safety assessments (stress tests) also indicate the need for measures in order to strengthen resilience against extreme natural phenomena, a loss of power and a loss of main heat sink. Furthermore, the plants' emergency preparedness and capability for emergency response management need to be strengthened in various respects. SSM will enjoin the licensees to carry out these safety improvements.

The accident at the Fukushima Dai-ichi nuclear power plant has also raised questions requiring more in-depth investigation and research to make it possible to draw conclusions about any additional measures at the nuclear power reactors. These areas are now being discussed in various contexts and research is being planned as part of international co-operation. SSM will take part in many of the investigations and research projects that are launched because of the accident.

SSM has made the assessment that the nuclear power reactors also need to be provided with systems for independent coolant makeup. This kind of system reduces the risk of core melt and thus also the risk of a melt-through of the reactor pressure vessel in the event of a loss of the ordinary coolant makeup system. SSM is now preparing requirements on independent coolant makeup systems and intends to impose requirements on the systems' design offering protection which, together with other protective measures at the facility, enable the additional coolant makeupsystem to be maintained to the extent needed in connection with a design basis threat covering the period of time determined by the Authority.

A review of the capability of licensees and the State to protect the facilities against antagonistic threats also indicates that protection against sabotage needs to be strengthened further. Investigations are in progress at SSM to ascertain which additional measures are needed, and amendments to the Authority's regulations in the area are under preparation.

Altogether, this implies that the nuclear power plants need to continue their work on analyses and measures at the plants to fulfil the requirements imposed by SSMFS 2008:17 with decided action plans as well as additional requirements on safety owing to the lessons learned from the Fukushima Dai-ichi accident, stress tests performed, safety investigations and investigations of physical protection. Continued safety improvement measures are also necessary for ongoing work to increase the safety margins against unforeseen events at ageing plants in long-term operation.

5.2. Ageing management and long-term operation

As with many plants around the world, when the Swedish nuclear power plants were designed and constructed, a period of operation of approximately 40 years was assumed. This included performing design analyses and fatigue calculations with assumptions concerning a certain number of startups and shutdowns of the facility, other operational modifications, scrams and various kinds of transients during this period of time. Consequently, long-term operation refers to operation beyond the period of time for which the facilities were originally designed and analysed. The licensees have announced that they intend to operate the nuclear power plants for 50 to 60 years. Swedish nuclear power plants are presently 27 to 40 years old, counting from the start of routine operation.

Long-term operation presents both licensees and regulatory authorities with new challenges, starting from the requirements imposed on safe operation during the extended period of operation. The organisation, resources and expertise of both the licensees and SSM must be adaptable to manage new safety issues that might arise in connection with long-term operation. Although many systems and components at the facilities have been replaced over the years in connection with safety upgrades or other refurbishing or rebuilding work and repairs, most crucial building structures, systems and components remain in their original design. The licensees' ageing management is for this reason a key area when it comes to safe long-term operation.

In its regulations, SSM imposes requirements on the licensees' ageing management work in terms of physical and technological ageing, and requires an ageing management programme for this area. An ageing management programme can be viewed as a comprehensive coordination programme consisting of other maintenance and inspection programmes, such as a surveillance programme for reactor pressure vessels, and programmes for environmental qualification, water chemistry and monitoring. Requirements are also imposed on the ageing management activities being subject to the licensees' management systems. The aim is to ensure long-term management of ageing issues and as far as possible preventing degradation and other deficiencies from arising so that barriers, structures and safety systems no longer work as intended. The requirements imposed on ageing management and ageing management programmes apply generally, but obviously increase in importance as the plants age.

The extensive research conducted nationally and internationally over the past 30 years or so has resulted in good knowledge about the ageing and degradation mechanisms that can give rise to damage at nuclear power plants. Consequently, these mechanisms can be satisfactorily handled with the inspection programmes and ageing management programmes applied today. Thus, these programmes should give good preconditions for safe operation, also in connection with long-term operation. There are, however, a number of areas in which ongoing inspections and analyses, in addition to development of methods and knowledge, are prerequisites so that these programmes can more effectively detect early indications suggesting safety deficiencies due to ageing over extended periods of operation. These programmes also need to be designed so that they as far as possible are capable of detecting completely unknown damage mechanisms and also known damage mechanisms that manifest themselves in unexpected places.

Prior to and during long-term operation, special attention also needs to be given to:

- irradiation embrittlement of reactor pressure vessels, particularly taking into account effects that can substantially increase the rate of embrittlement
- fatigue, particularly taking into account impact from the reactor water environment on areas sensitive to fatigue
- the condition of tendons and steel liners in reactor containments
- degradation mechanisms that can influence reactor containments' concrete and metal parts
- possibilities for reliable inspections and testing of reactor containments
- the validity of environmental qualifications of electrical, instrumentation and control equipment as well as parts with polymer construction materials

Ongoing knowledge building is necessary for the long-term application of effective inspection programmes in terms of stress corrosion in: a.) components manufactured of nuclear grade material, and b.) certain components in pressurised water reactor environments.

As far as concerns even longer periods of operation, approaching 60 years, continued investigation and research are needed for timely detection of degradation due to fracture toughness deterioration as a consequence of thermal ageing of stainless steel welds and cast stainless steel.

Thus, what is crucial as to whether a reactor can be operated further over extended periods and with a sustained level of safety is the licensee applying a thorough and effective ageing management programme. In the present analysis of ageing issues in connection with long-term operation, SSM has pointed out a large number of measures that need to be taken prior to adopting a position on such operation, while also highlighting the Authority's point of view on necessary parts of the licensees' ageing management work.

Using this as a point of departure, SSM intends to adopt a standpoint on long-term operation of nuclear power plants on the basis of periodic safety reviews under the requirements imposed by the Act on Nuclear Activities and the Authority's regulations. Clarification and more precise wording of SSM's regulations and general advice concerning periodic safety reviews are being planned in terms of (for example) the aspects described above that are of importance in connection with long-term operation. SSM has nevertheless already established that an account of a licensee's periodic safety review that is to serve as the basis of the Authority's standpoint on long-term operation needs to, for example, encompass analyses describing the facility's ageing status over time for certain key parameters, such as irradiation embrittlement of reactor vessels, component fatigue and tensioning force loss in the reactor containment. This also applies to analyses and when checking the condition of safety-critical electrical cables, instrumentation and control equipment.

5.3. Need for changed regulation and regulatory supervision

SSM currently has a satisfactory model for regulation and regulatory supervision in the field of reactor safety. This model stands up relatively well in relation to international standards and practice, but it needs to be developed in various respects. The model mainly represents regulation and regulatory supervision focusing on licensees' management and control of their activities and has evolved over the past 20 years, primarily within SKI, the former regulatory authority. The principles for this model were established in a situation where all Swedish nuclear power plants were to have been shut down by 2010. Among other things, this situation implied that the focus of the regulatory model came to be placed on safety issues related to operation and maintenance. Part of this model includes a general kind of regulation imposing generally worded requirements on the work activities at Swedish nuclear power plants, also with supervisory work oriented at the licensees' management, control and follow-ups of the organisation's work.

The results of the IRRS review and modifications to nuclear activities in Sweden, such as extensive safety upgrades at the facilities, planned long-term operation plus increased international regulatory collaboration and a higher level of harmonisation of nuclear safety regulatory supervision, show that the model applied needs to be developed and made more precise. The present analysis shows that this further development and modification needs to encompass not only the regulatory framework, in the form of regulations and general advice, but also regulatory supervision.

SSM's regulatory framework needs to become more comprehensive and be based on international safety standards and European practice, while also demonstrating improved predictability about the implications of the requirements imposed. SSM is now preparing this regulatory revision. The challenge for SSM will be to supplement and clarify regulations and general advice concerning nuclear power safety to enable achievement of these objectives while not going too far in terms of detailed regulation in a way that leads to unclear circumstances about a licensee's safety responsibility. Key points of departure as part of the regulatory framework's revision will include the IAEA's new safety standards and the 'safety reference levels' and other documents agreed between nuclear regulatory authorities in the European Union within the WENRA collaboration.

As up until now, and in agreement with the Government's communication to the Riksdag, regulatory supervision is to have its foundation on the controlling element in the supervision, but with an orientation and execution that are more adapted to the varying nature of the matters of supervisory work. The objective for the further development and this change is for each situation having a correct orientation of regulatory supervision that is pursued in a way that is fit for purpose and effective.

SSM will consequently, at the same time as the revision of the regulatory framework takes place, be refining strategies and approaches for conducting regulatory supervision in the field of reactor safety. In order to achieve a regulatory supervision that is more effective and even more appropriate for purpose, approaches and strategies need to be developed for various areas of regulatory supervision and be adapted to the nature of the areas and the matters of regulatory supervision as well as their importance for safety. The starting point for this work will be the findings of the present studies within international co-operation concerning regulatory approaches and strategies.

This means that SSM will be defining the areas of supervision for which different supervisory strategies are to be developed. For each area, it is defined which combinations of approaches are to be applied and serve as the basis for the strategy per area. This strategy is also to be designed so as to facilitate application of the graded approach. In certain areas, consideration will also be given to recommendations from the IRRS review mission, as well as recommendations from public sector collaboration as part of CNRA, WENRA and ENSREG.

This for example applies to periodic safety reviews of the facilities that are to be conducted at least every ten years. The requirements imposed on these kinds of safety reviews have for a long time now been stipulated in the authority's regulations, but were incorporated in the Act on Nuclear Activities in 2010 on the justification that these reviews are of principal importance. Even internationally, these kinds of periodic safety reviews (PSR) are of key significance for ensuring that operational experience, new knowledge and new safety standards are taken into consideration and have an impact at the facilities. Against the background of the Fukushima Dai-ichi accident, it may also be anticipated that PSRs will be attributed greater importance internationally in the regulatory authorities' work on regularly evaluating the validity of the applied design bases, assumptions and safety analyses. SSM will monitor this international development and adapt the application of the periodic safety reviews thereafter. SSM also intends to give the periodic safety reviews a formal role when it comes to adopting a position on long-term operation of Swedish nuclear power plants in accordance with recommendations from CNRA and other bodies.

Other areas that have been emphasised by the IAEA, as part of public sector collaboration as well as by the present report, and which will be encompassed by SSM's revision of its regulatory model, include:

- supervision of ageing management during long-term operation
- supervision of the licensees' control and quality assurance of work performed by suppliers and contractors
- analyses and follow-ups of operating experience

SSM will also continue with development work initiated at the Authority, including annual integrated safety assessments of the nuclear power plants and a more extensive follow-up of safety culture matters.

These changes to the regulatory framework and regulatory supervision will require more resources at SSM. This applies both to work on changes to the regulatory framework and to the Authority's regulatory supervision, and to the work on examining periodic safety reviews of facilities. The IRRS review also drew the conclusion that SSM needs increased resources for meeting the challenges faced by the Authority.

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