R&D Arena Barsebäck

Feasibility study and coordination proposal

Elforsk rapport 14:56

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ELFORSK

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Förord

Denna förstudie har genomförts inom ramen för Elforsks kärnkrafttekniska forskningsprogram och initierades vid ett möte hos E.ON i Malmö i februari 2014 där ett 15-tal deltagare diskuterade möjliga forskningsfrågor som kan utredas i Barsebäck, och vad som behövs för att föreslå ett sådant FoU-program.

Förstudien utfördes av Hans Henriksson, Vattenfall R&D AB under 2014 i nära samarbete med representanter i olika Elforskprogram samt med forskningsansvariga vid kärnkraftverken.

En referensgrupp bildades och har träffats två gånger. Referensgruppen bestod under denna etapp av Monika Adsten (Elforsk), Tommy Holm (OKG), Henric Lidberg (RAB), Bengt Sikland (BKAB) och Jan-Anders Svensson (E.ON).

Preface

This feasibility study has been financed by the Elforsk Nuclear Energy Research Program, and was initiated at a workshop at E.ON in February 2014 where about 15 participants discussed possible R&D activities at Barsebäck, and the continued actions to propose such R&D programme.

The study has been conducted by Hans Henriksson, Vattenfall R&D AB during 2014 in close cooperation with representatives in the different Elforsk programmes as well as the research responsible at the nuclear power plants.

A reference group was formed and met two times. In this phase it consisted of Monika Adsten (Elforsk), Tommy Holm (OKG), Henric Lidberg (RAB), Bengt Sikland (BKAB) and Jan-Anders Svensson (E.ON).

Stakeholders of the Elforsk Nuclear Energy Research program:

Sammanfattning

Denna förstudie har som övergripande mål att sammanfatta intresset för Barsebäck som en forsknings- och utvecklings- (FoU) Arena för åldrings-, säkerhets- och driftsrelaterade studier. En lista med potentiella experiment och projekt presenteras, med en övergripande tids- och kostnadsplan. Ett tydligt behov av en koordinering av denna FoU arena har identifierats och därför föreslås att ett forskningsprogram startas inom Elforsk, där representanter från kärnkraftverk, kraftverksägare och säkerhetsmyndighet ingår. Granskning av projektförslag kommer att ske i redan etablerade expertgrupperingar, såsom Elforsks betongprogram, materialgruppen och beräkningsgruppen (där FKA, OKG, RAB och TVO ingår).

Nästa steg är att ta fram en kortlista innehållandes föreslagna forskningsaktiviteter med fokus på 1-2 pilotprojekt som startas under 2015.

Arbetet med denna förstudie har genomförts genom skriftliga och muntliga utfrågningar, möten med huvudintressenter såsom Elforsks betongprogram-styrgrupp, Chalmers åldringsdag inom SNEC (centrum för hållbar kärnenergi) och med Älvkarlebylaboratoriet inom Vattenfall R&D. En lista på referenser och kontaktpersoner finns med i rapporten.

Hans Henriksson

Oktober 2014

Summary

This feasibility study has the overall goal to summarise the main interest for Barsebäck as an R&D Arena on ageing mechanisms and safety demonstration studies. A list of potential experiments and studies is presented, with an estimate on time and cost. A clear need for a coordination group is identified, with the proposal that an R&D Programme under Elforsk is formed, consisting of representatives from the main stake holders: nuclear operators, utilities and safety regulators. The review of project proposals will be carried out in the already established networks of experts: Elforsk concrete research programmes, Materialgruppen and Beräkningsgruppen (organised between FKA, OKG, RAB and TVO).

The next step in the forming of R&D Arena Barsebäck is to propose a short-list of suggested activities with emphasis on one or two pilot projects to be launched in early 2015.

The work in this feasibility study has been conducted through questionnaires, meetings with the main stake holders, such as at the Elforsk concrete research programme steering group, Chalmers Sustainable Nuclear Energy Centre (SNEC) ageing day, and with Älvkarleby Laboratory, Vattenfall R&D. A list of references and contacts can be found in the report.

Hans Henriksson

October 2014

Innehåll

Acronyms

AMP	ageing management programme
BKAB	Barsebäck Kraft AB
BWR	boiling water reactor
CRP	IAEA organised coordinated research project
DoE	U.S. Department of Energy
EAF	environmentally assisted fatigue
EPRI	Electric Power Research Institute
FEM	finite element methods
FKA	Forsmarks Kraftgrupp AB
FSI	fluid structure interaction
IAEA	International Atomic Energy Agency
IASCC	irradiation-assisted stress corrosion cracking
I&C	instrumentation and control
IGALL	international generic ageing lessons learned
LTE	life-time extention
LTO	long-term operation
LWR	light water reactor
MAaD	materials ageing and degradation
MAI	Material ageing institute (EDF, France)
NBSG	National Fire Safety Group, nationella brandsäkerhetsgruppen
MG	Materialgruppen
NDE	non-destructive examination
NEA	OECD Nuclear Energy Agency
NKS	Nordic Nuclear Safety Research
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
OECD	Organisation for Economic Co-operation and Development
OKG	Oskarshamns Kraftgrupp
PSA	Probabilistic Safety Assessment
PWR	pressurized water reactor
R&D	research and development
RAB	Ringhals AB
RCB	Reactor Containment Building
RPV	reactor pressure vessel
SALTO	safety aspects of long-term operation
SCC	stress corrosion cracking
SKB	Nuclear Fuel and Waste Management Company
SS	stainless steel
SSM	Swedish Radiation Safety Authority
TH	thermal hydraulics
TLAA	time limited ageing analysis
TVO	Teollisuuden Voima Oyj
VRD	Vattenfall Research and Development AB
ZIRP	Zorita Internals Research Project

1 Background to this feasibility study

The operating time of the Swedish nuclear power plants will be increased from the original 40 years to at least 50 years. In order to insure a safe operation, ageing effects must be investigated through tests, modelling and verification. Destructive testing and investigations causing, or being associated with, risk for substantial unavailability can only be carried out to some extent in operating power plants and are more suitable for plants that are no longer in operation.

The Barsebäck reactors were transferred to service operation in 1999 and 2005, respectively. Awaiting the decommissioning, a business function has been started working with education, selling components, research etc. The reactors are though basically intact, and it is possible to use the facility for research and tests at least until 2021, as long as it avoids disturbing the training and education activities and the planned removal of the internals in 2016-2018.

Elforsk coordinates some Research and Development (R&D) for the Swedish utilities and nuclear regulator. The company is coordinating research activities within the field of nuclear in the areas of concrete structures, polymers, I&C and vibrations. There are also a number of collaboration groups outside Elforsk working with nuclear research financed by the Swedish nuclear power plant owners, for example within metallic materials, calculations, etc.

Fig. 1. Barsebäck 1 and 2 (Courtesy BKAB)

1.1 Workshop

E.ON in cooperation with Elforsk initiated a workshop on February 11th 2014 to discuss the possibilities of using Barsebäck as an R&D arena, taking samples and performing tests, before it is decommissioned. The workshop gathered some 15 representatives from the Swedish and Finnish nuclear power plants and from the owner companies of the plants. During the discussions three possible areas of research/testing were identified: operational safety, ageing and decommissioning. It is vital that the activities are coordinated in order to secure that they do not destroy the prerequisites for coming tests and final decommissioning. The notes from this workshop are found in Appendix A.

1.2 Coordination

An R&D arena should be coordinated through already existing cooperation, such as the Elforsk concrete research program and the cooperation group for metallic materials (Materialgruppen, MG) should be used in the planning of the research program to avoid forming parallel groups. During the discussions the focus was on the areas ageing and operation, since large resources are already put into the area of decommissioning in other forums. A team work was performed where the participants were divided into three groups discussing the need for research and testing within the areas metallic materials, concrete structures and operation.

The major need is prioritization and coordination between all participants and stakeholders. This should mainly be carried out by Elforsk. Elforsk could also contribute with the overall connection between areas of expertise, as well as the coordination of partners and funding. However, it is important that Barsebäck informs about their plans and discontinue project proposals that could alter other activities on site.

Stakeholders include Nordic nuclear power plants (NPPs), Westinghouse Sweden, Nordic Owners Group (NOG), Swedish Centre for Nuclear Technology (SKC), BWR Club, IAEA and NEA ageing projects, EPRI (BWR VIP, MRP), NUGENIA, ASME, IEC/IEEE. A questionnaire on requested activities in R&D Arena Barsebäck (see Appendix B) was sent out to many of these parties.

1.3 Barsebäck units 1 and 2

The Barsebäck 1 and 2 NPPs (see Fig 1) were ordered from Asea Atom in 1969-70 with the nominal output power of 570 MW. Turbines and generators were ordered from Stal-Laval. Barsebäck 1 was loaded in 1975 with first criticality Jan 18. From May 15 B1 could deliver power. B2 delivered power in March 1977 and run on full power from June 1977.

E.ON Kärnkraft Sweden owns the Barsebäck plants. The present operation of Barsebäck is carried out by Barsebäck Kraft AB (BKAB) owned by Ringhals AB (RAB), which is owned by Vattenfall and E.ON. An aerial photo can serve as a plant overview of Barsebäck, see Fig. 2.

Fig. 2. Barsebäck plant overview: (1) reactor buildings, (2) turbines, 3 offices, (4) Filtra unit, (5) cooling water inlet, (6) water filtering, (7) Waste disposal. Courtesy BKAB.

The turbines consist of one high-pressure and three low-pressure turbines on the same axis as the generator. The two Barsebäck units were constructed by Asea Atom, with turbines from Stal-Laval. An overview of the main features of the two Barsebäck plants is given in ***Table 1*** below. See also Appendix C for further details on the Barsebäck plant [BKAB].

Table 1. Basic facts about the Barsebäck 1 and 2 plants (same values for both units).

Thermal reactor power, MW	1800
Output power, MW	570-615
Number of circulation loops	4
Operational pressure, bar	70
Total coolant water flow, kg/s	7700
Steam flow, kg/s	900
Steam pressure, MPa	7,0
Steam temperature, °C	286
Feedwater temperature, °C	184

2 Scope and objectives

The interest in ageing studies for long-term operation (LTO) and Life-Time Extension (LTE) of nuclear power plants has been important the last years. Nuclear power plants around the world, including the Swedish plants, were designed and constructed for an assumed period of operation of approximately 40 years. Consequently, LTO 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 28 to 42 years old, counting from the start of routine operation.

The Swedish Radiation Safety Authority (SSM) indicates that for LTO, special attention needs to be given to [SSM2010]:

- irradiation embrittlement of reactor pressure vessels (RPVs), 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 (I&C) equipment as well as parts with polymer construction materials

Ongoing knowledge building is necessary for the long-term application of effective inspection programmes for stress corrosion cracking (SCC) in:

a) components manufactured of nuclear grade material, and
b) certain components in pressurised water reactor (PWR) 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 (SS) welds and cast SS.
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 (AMP). 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.

SSM has already established that an account of a licensee's periodic safety review 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.

SSM also carried out an investigation on some specific areas of interest in 2012 [SSM2012] to give their standpoint on what requirements are reasonable to demand from the operators:

- Fatigue studies including transients
- RPV embrittlement
- Loss of tension in reactor containment building (RCB) tendons
- Specific analyses, for example cracks under RPV cladding of Ringhals 2

As a result of the enlarged interest for these questions, R&D Arena Barsebäck is proposed and detailed below.

2.1 Benefit of R&D activities in Barsebäck

Several studies and investigations related to ageing as well as safety aspects on LTO (SALTO) can be carried out in operating reactors of today, which is already part of the general safety review during normal operation. However, some studies are not feasible to carry out, such as destructive testing of components or structures, tests outside safety margins and design limits, validation of safety functions linked to severe accidents, and full-scale tests of entire systems of components. These tests are unfortunately often time-consuming, costly and complex by nature, and a detailed experiment planning is needed.

The possibilities in Barsebäck to carry out such studies exemplified above, and proposed below, are important and should be used in an optimal and timely way so that this opportunity is not lost after the decommissioning.

The R&D arena offers means to provide conditioned open access to projects to be carried out. A model-based approach is favoured where an established model of the reactor containments would be the basis for the planning of experiments and tests, which in turn can be used to update the models. This way of working enables different research groups to, over time, contribute to the knowledge on how to establish models as part of the AMP of operating power plants.

It should be noted that there are major similarities with Barsebäck 1 and 2 and most other Nordic BWRs. B1, B2 and Oskarshamn 2 are called the triplets, indicating identical layout and design.

2.2 Restrictions in using Barsebäck

It should be noted that the proposed activities within this programme need to be well in line with other activities on-site:

- Limit risk of disturbing training sessions, such as KSU activities,
- Avoid activities that could interfere with the removal of internals
- No activities that could cause changes or delay in the final decommissioning and dismantling.

BKAB will have the final decision on a proposed R&D activity. However, even destructive testing can be performed, such as complete validation of non-destructive examination (NDE) techniques.

2.3 Focus areas for the R&D Arena

According to the IAEA Safety guides several issues need to be addressed concerning ageing [IAEA2009]:

In order to manage ageing processes the programme should include, but should not be limited to, the following aspects:

- *identification of components important to safety that are susceptible to degradation through ageing,*
- *identification of degradation processes that could adversely affect plant safety,*
- *adequate and up to date methods for detecting and monitoring ageing processes,*
- *the keeping of appropriate records to enable the ageing process to be tracked,*
- *methods of taking corrective actions in order to mitigate and/or prevent the effects of ageing,*

The identified areas for consideration in this programme, which also links to AMPs internationally reported in the “international generic ageing lessons learned” (IGALL) report from the IAEA [IAEA2010], are:

- Operation and safety
  - Validation of thermal hydraulics codes and pressure gradients
  - Verification of NDE techniques
  - Fire studies
- Ageing
  - Containment and civil works
  - Internals, RPV and cladding
  - Polymers

These areas are discussed in section 3-5, with a summary of proposals in the conclusions (section 7). Appendix D lists all potential activities proposed to date, and Appendix E exemplifies some of these proposals more in detail.

2.4 Other areas considered

Beside the areas mentioned in section 2.3 above, there are also other interests in experiments to be performed at Barsebäck. Studies on decontamination techniques as well as dismantling procedures could be undertaken as well. Rivningsgruppen and the Nuclear Fuel and Waste Management Company (SKB) should be involved. Examples of other studies that are carried out at Barsebäck are listed below:

- Defects in concrete – carried out by Scanscot
- Defects in concrete reinforcement – Force Technology
- Irradiation degraded cables – Chalmers
- Radiation testing of cameras inside the RPV – Ahlberg Electronics
- Material testing of core barrel specimens – Vattenfall
- Material testing of RPV surveillance specimens – Vattenfall
- Long-term testing of pumps on long shafts – OKG

Duplication of work is to be avoided, which means that the decision on new R&D tasks should be based on the outcome and results from previous and parallel ongoing work

3 R&D Area "operation and safety"

The Barsebäck units are well suited for a variety of activities linked to improvement and validation of operability and safety margins. This part is split into issues related to Instrumentation and Control (I&C), Thermal Hydraulics (TH), and Fire. Other areas of interest linking to these three issues may be found in Section 4 (ageing). A comprehensive list of project proposals can be found in Appendix D.

3.1 Instrumentation and Control (I&C)

This topic is often depending on site-specific I&C details which means a limited use of results from activities carried out at Barsebäck. However, in some areas, there are important contributions from such undertakings, such as connection to ground or equipment behaviour in harsh environments (radiation, heat, humidity).

The main expert group for coordination of Barsebäck R&D in the area is the Elforsk I&C Programme. Some of the proposals discuss related issues, such as relay functions, and the electric ground gridline degradation. Corrosion issues are substantial, such as electrical stray current corrosion. See also Appendix E for further details.

3.2 Thermal hydraulics

Typical issues involve validation of codes relevant for studies of pressure transients through valves and other components, fluid-structure interaction (FSI) and loads on structures. Another area of interest is stratification. The possibility to carry out full-scale TH experiments at plant level is what is unique to Barsebäck.

Expert groups for coordination and project review include DKC-TS (Vattenfall-FKA-RAB network on TH issues) and Beräkningsgruppen (FKA, OKG, RAB and TVO collaboration).

The project proposals first includes a pressure transient study with a sudden change in pressure through a pipe system due to rapid closure of a valve (i.e. a water hammer), which would be used for validation of codes used for load determination as well as safety assessment studies.

Another proposal suggests analysing stratification in the pressure suppression pool. This is also of interest to study at full-scale in order to get rid of scale effects in a lab environment and is relevant for the validation and development of codes. See also section 4.1 (Concrete) regarding further projects related to TH studies as well as Appendix E for further details on some proposals. A list of some experiments is given in Table 2 below.

Table 2. List of project proposals related to thermal hydraulics.

Project group	**Comment**
Pressure transients	*Medium-term projects with high priority*
Water hammer study	Common needs at all NPPs
Flooding and loss of coolant	Validating model of simulated break
Pipe breaks	
Dynamics in pressure suppression pool	*Short-term projects*
Stratification and temperature mixing	Full-scale experiment being unique
Validation of codes	Benchmark study. Comparison with previous test at Forsmark 1.
Test of FILTRA	*Medium-term projects*
Study of filtering capacity	Accident scenario studies

3.3 Fire

Several studies include fire as well as ventilation and escape routes for smoke and toxic gases. Several experts are coordinated within the National Fire Safety Group (NBSG) were participants from Swedish Radiation Safety Authority (SSM), SKB and Swedish nuclear operators are members. The scope of NBSG is to find synergies in the area of fire safety through a common financing of research, tests information exchange etc. The following areas are included in the activities of NBSG:

- Give a common view from operators to the organization BRANDFORSK
- Carry out R&D for Fire Safety in Swedish nuclear industry
- International Outlook and interpretation of rules and regulations
- Support and follow ongoing research projects.

The interface and information exchange with other groups in the Fire Safety area is guaranteed by the representation in BRANDFORSK and within the Nordic collaboration for Probabilistic Safety Assessment (PSA) called NPSAG. Safety studies within the Nordic Nuclear Safety Research (NKS) could also be considered, as a forum for cooperation and competence in nuclear safety with joint activities financed and supported by Nordic authorities, companies and other organizations.

Table 3. List of project proposals related to fire safety studies

Project group	**Comment**
Fire projects	*Long-term projects*
Flow rate in chimney	Common needs at all NPPs
Smoke and gas diffusion in building	Model validation
Real fire experiment in confined area	Test in relay station or turbine
Test of fires in battery backup	Risk assessment of large battery banks

4 R&D area “ageing”

The area of material ageing and degradation (MAaD) is of main interest for nuclear LTO studies and in support of present AMPs with the added value of cost reduction for continued operation of today’s nuclear power plants. It should also be put in perspective of the formal requirements by the regulator. A formal process for approving LTO is needed in order to demonstrate safe operation. For countries with no time-limiting licenses (such as Sweden), usually a periodic safety review (PSR) process is used. SSM will use the review of the ageing management programs and PSRs to endorse LTO. The PSR requirements are included in the Act on Nuclear Activities (1984:3) [SKI1984] where it says that an overall assessment of the plant safety and radiation protection shall be performed at least every 10 years. The assessment shall account for the R&D and progress in science and technology.

From the time limited ageing analyses (TLAA) listed in Ref [SSM2012] some examples are given for continued focus:

- Embrittlement of the RPV
- Fatigue and transient monitoring
- Loss of containment tendon prestress,
- Under clad cracks in RPV

The US Nuclear Regulatory Commission (NRC) also states that [NRC2010] potential TLAAs include flow-induced vibration endurance limit for internals, transient cycle count assumptions for internals, leak-before break, fatigue analysis for containment liner, metal corrosion allowance and low-temperature over-pressure protection analyses.

4.1 Concrete

There are several recent or on-going projects in this area, such as Elforsk studies on air-ingress and PhD studies on climate conditions inside the containment and the project CONSAFESYS (Concrete Containment Condition Status & Ageing Examination System) with the objective to develop NDE methods combined with Finite Element Methods (FEM), with several Swedish participants. Scanscot, SP, SSM, OKG, Fortum, Ringhals are involved among others. Another important project was the “Concrete containment management using the finite element technique combined with in-situ non-destructive testing of conformity with respect to design and construction quality” (CONMOD), where Scanscot, Force, EDF and BKAB were partners.

The expert group for coordination would be the Elforsk concrete research programme, with participants from Vattenfall, FKA, RAB, OKG, SSM and TVO. Below follows a list of potential projects related to civil works MAaD studies with comments. The complete list of project proposals can be found in Appendix D.

Table 4. List of project proposals related to ageing studies of concrete and civil works.

Project group	Comment
Fuel pool studies	*Long-term projects to be performed close to dismantling*
Leak test at boiling	*Common needs at all NPPs*
Validation studies of models	*Links to the safety area (see above)*
Deformation study	
Tendon and rebar studies	*Short term project of high priority*
Relaxation after plasticisation	*Barsebäck has grouted tendons, similar to R1, 01-O3 and Olkiluoto 1.*
Long-term testing	*Validation of NDE and verification of actual status of reinforcement.*
Corrosion studies	*Links to ongoing activity in ConSafeSys (and previous ConMOD)*
Normal reinforcement condition	*Interest from SSM on these issues.*
Water-way study in dry condition	Verify if Barsebäck is too "unique". Corrosion issues are substantial, such as electrical stray current corrosion.
Crack systematics and liner condition	*Medium-term study*
Surface cracks in liner	*Needs further specifications.*
Validation of NDE methods	*Could be costly for a systematic approach but very valuable*
Verification with samples	SCC studies
Study of cracks in rotor blades	
Polymers in concrete	*Short-term projects*
Study of polymers in concrete	*Need further proposal investigation.*
Lubricants	*Links and similarity to other NPPs?*
Chemical anchors	*Should be discussed in the Elforsk Polymer programme.*
Cable penetrations and sealing	
Surface protection	*Long-term projects*
Paint ageing	*Further need of project objectives.*
Paint fall-off during depressurization	*Could be carried out in connection to overpressure and leak-tests.*

Project group	Comment
Containment over-pressure studies	*<u>Long-term safety study</u>*
Air-ingress in concrete	*Pressure above design margins would be beneficial for safety cases.*
Containment deformation	*Scanscot carries out activities related to precipitates.*
Monitoring of liner deformation	*International benchmarking possible.*
Deformation after depressurisation	
Irradiated concrete studies	*<u>Medium-term study</u>*
Containment specimen testing	*OKG are already active in this area.*
Microscopy of concrete cores	*These studies link to the humidity analysis.*
Humidity level in containment	*<u>Short-term study</u>*
Monitoring humidity in biological wall	*Links to ongoing studies carried out at several NPPs in Sweden.*
Humidity distribution in outer wall	*Can be started quickly.*
Loads of blow-down experiments	*<u>Long-term study</u>*
Vibration in critical locations	*Links to Marviken studies in the 1970s.*
Pressure transient thermal hydraulics study	*Project links to pressure transient studies under “Safety studies” above.*
Impact load studies	*<u>Long-term project close to dismantling</u>*
Induced vibrations due to external source	*Is carried out with external impact (hammer) on containment.*
International benchmarking of impact load	*Benchmark on code validation.*

4.2 Metallic materials

Material testing is destructive by nature. Therefore, the need for coordination and prioritization is clear. The main proposal on metallic analysis is to take samples from components situated in a harsh environment, such as internals and the RPV, including SS and welds. *Fig. 3* shows a photo of the RPV installation at Barsebäck during the construction phase.

The main expert group involved in reviewing experimental proposals is Materialgruppen with representatives from FKA, OKG and RAB.

Fig. 3. RPV during plant construction (Courtesy BKAB).

Below follows a list of potential projects related to metal MAaD studies. The complete list of project proposals can be found in Appendix D.

Table 5. List of project proposals related to metal ageing studies

Project group	Comment
Tendon studies	<u>*Short term project of high priority*</u>
Relaxation after plasticisation	*(See concrete studies above)*
Long-term testing and corrosion	
Normal reinforcement condition	Corrosion in re-heater and heat exchangers
Crack systematics and liner condition	<u>*Medium-term study*</u>
Surface cracks in liner	*(See concrete studies above)*
Validation of NDE methods	SCC studies such as the irradiation-assisted SCC (IASCC)
Verification with samples	
Study of cracks in rotor blades	
RPV embrittlement analysis	<u>*Short-term study*</u>
Irradiated specimen testing	Ongoing plans for surveillance testing of B2 specimens
Material samples (trepan)	Test of attenuation effects
RPV liner weld studies	Microscopic studies of fracture interaction

Project group	Comment
Stainless steel studies	*Medium-term study*
Valve housing and internal structures	Study of component ageing trough
Study of Stellite and surface material	Crack systematics
Welds in Ni-based alloys and SS	Study of crack propagation in core spray system, and analysis of alloys 600, 82, 182, X750.
Electric ground connection	*Long-term study*
Study of ground gridline degradation	Similar solution at all NPPs. Could be tested during dismantling.

4.3 Polymers

Areas of special interest for further studies are where components need extended maintenance (costly and time-demanding), such as for sealing strips, bellows and glands in the RCB, see an example in *Fig. 4*. Also, coating materials are of great interest to study as that is complicated and costly to replace. It should be noted that polymer material choices within the RCB were well considered at the time of construction.

Fig. 4. Sealing strip from Ringhals after 30 years of operation [ELFORSK2013].

There is a need for transparent and clear criteria of when a component is exhausted in terms of life-time. By analysing components that have been in operation and suffered from environmentally assisted fatigue (EAF) the competence within the company can be maintained. The estimate of life-time could be reassessed depending on functional requirements as well.

Long-term testing in a component's environment should be applied. Many phenomena (e.g., degradation of antioxidants or different diffusion rates) occur at lower temperatures. In order to gain a deeper understanding of all mechanisms and therefore also to optimize the material's life time, such approach is needed although this needs to be compared with results from the accelerated MAaD.

Materials that have been operating in a known environment and aged for many years is invaluable when it comes to increasing the understanding. Examination of such material leads to a deeper insight of mechanisms that have aged the material. The result also contributes to a better lifetime estimate.

For some components, such as coatings, a number of other approaches should be used. Coatings of epoxy are problematic to replace; would it be possible to re-coat existing coating or must it be removed? What would the cost be while maintaining a clean system? Are staff trained on this topic and are today's coatings approved with respect to present chemical rules?

Some examples mentioned on proposed projects for further study:

- Leakage through polymer sealing strips in the RCB.
- Studies on cable penetrations (so called ”Brattbergare”) and rubber bellows (such as EPDM) to find mechanisms limiting component lifetime and maintenance/replacement scheme.
- Sealant ageing with criteria and tests for the decision on actual status.
- Coating and maintenance of coated surfaces. Do the replaced coatings function differently than before?

The Elforsk polymer network is suggested to review projects in this area. Below follows a list of potential projects related to polymer MAaD studies. The complete list of project proposals can be found in Appendix D.

Table 6. List of project proposals related to polymer ageing studies

Project group	**Comment**
Code validation and comparison between B1 and B2	*Medium-term project*
Irradiation effects	Limited importance as B1 and B2 have almost the same operating age.
Ageing in different environments	Study of EAF phenomena of high priority.
Condition of sensors and switches	Replacement strategy comparison.
Interaction with concrete	*Medium term project*
Anchor plate studies	EAF study and links to international collaboration.
Lubricants used in containment	
Sealing and cable feed-through	
Surface protection (e.g. paint)	

5 Other areas of interest

The main focus on ageing degradation and safety-related studies cover most aspects that could be worth studying. However, it would be reasonable to also investigate other needs of Barsebäck within the nuclear area. In this section some proposals are listed that go outside the general scope of this R&D Arena work.

5.1 Decommissioning

The main expert groups involved in decommissioning are coordinated in Rivningsgruppen, with SKB as organiser. These experts review normally all R&D-related activities in this area. However, it has been noted that some issues are not yet covered in present studies, such as procedures and techniques on the dismantling. Barsebäck could be a full-scale test facility for comparing different techniques in two almost identical units. One such test could be the comparison of wet-cutting or plasma cutting.

5.2 Other potential topics

A few other issues could be considered in a wide study, such as material studies or replacements of components for future nuclear reactors. This could be validation of new components through comparative studies with old well-proven components is also of great value.

Other studies also include deepened plant safety studies, project management and maintenance improvement.

6 Studies carried out elsewhere

6.1 Swedish studies

6.1.1 Marviken steam explosions and pressure suppression function

In the early 1970s, Sweden had a complete nuclear power plant ready for operation. It was however never started. Instead, it become a full-scale test site for a number of important experiments to improve nuclear safety world-wide. Experiments include steam explosions in the RCB as well as the pressure-suppression principle. About 60 different experiments were carried out from 1972 to 1986, including loss-of-coolant accidents (LOCA) and aerosol transport simulating fission product release after a core meltdown. A large number of countries in Europe as well as USA, Canada and Japan financed the experiments [Thorén]. See also Appendix F for further details. The photo below (*Fig. 5*) was published in a local newspaper at the time of the test.

Fig. 5. Simulated steam explosion at Marviken. The steam coming out of the reactor building fell later down as a luke-warm rain. Photo by Ingvar Anehed [NT2011].

6.1.2 Chalmers ageing study

An ageing report entitled “Ageing of components in nuclear power plants”, [Liljentzin2013] was recently presented and as a follow-up a workshop on ageing was held in May 2014, organised by the Sustainable Nuclear Energy Centre (SNEC) at Chalmers. The workshop gathered about 40 participants from the Nordic operators, utilities and research institutes. Conclusions from the ageing report are included in Appendix D.

6.2 International Ageing activities

6.2.1 IAEA and NEA

Examples of previous studies carried out are the projects at the Greifswald WWER-440 plant shutdown in 1990. RPV trepans were extracted from for example the circumferential beltline weld seam to see radiation response of the overlay cladding and other RPV material investigations [Rindelhardt2009].

A coordinated research project (CRP) on the “evaluation of properties and component materials utilizing actual aged materials removed from decommissioned reactors for LTO” has been initiated recently with the focus on metal- and civil works ageing. This CRP will be of importance to study and Barsebäck experiments could be of value to the CRP as well.

A number of workshops on Non-destructive Evaluation of Thick-walled Concrete Structures (WGIAGE) have been organised within the NEA. The objectives are to review existing methods for non-destructive evaluation of NPPs thick-walled concrete structures and to present state-of-the-art techniques for the integrity of concrete structures, aiming to detect voids and cracks in concrete and to detect passive (rebars) and active (tendons) reinforcing items.

The decommissioned Spanish NPP Zorita (or Jose Cabrera with about 26 full-power years of operation) is being used in the Zorita Internals Research Project (ZIRP) with the objective to fill existing knowledge gaps and improve understanding of the evaluation of core internals in both BWRs and PWRs, related to mechanical (tensile strength, fracture toughness, crack initiation and growth) and microscopic properties (grain boundary chemistry and size, void formation, and hydrogen and helium production). Results foreseen are to be used for licensing purposes, inspection programmes, ageing management and lifetime extension. The US Electric Power Research Institute (EPRI) is involved in ZIRP as well as NRC, Tractebel, Ringhals, SSM among others.

Studies in USA

Work funded and managed by EPRI under their LTO Program is organized and managed in the following work areas:

1. Primary System Metals Ageing
2. Concrete Structures, including Containment Degradation
3. Instrumentation and Control (I&C) and Information Technology (including online monitoring of critical equipment)
4. Advanced Safety and Risk Analysis Tools

5. Integrated Life-Cycle Management (data, methods, and tools)
6. Cable Ageing
7. Ageing Management Program Scope for Operation Beyond 60 Years
8. Integrated Strategy, Process Plan, and Demonstration Plants.

Concerning material studies, both the BWR Vessel and Internals Project (BWR-VIP) and the Materials Reliability Program (MRP) is of importance to LTO. Other projects within EPRI include water chemistry and NDE.

Containment integrity research has been performed in for example Sandia National Lab, with an extensive list of experiments since 1973. Some of these are listed in *Table 7* below. An example of such RCB failure test was performed in 2001, see *Fig. 6*.

Table 7. List of some containment integrity reseach at Sandia Nat. Lab. [SANDIA2006]

1973 – ASME B&PV Code, Section III, Div. 2 (ACI-359) (concrete containment design rules)
1982 – NRC-Sponsored Containment Integrity Program at SNL
1982 – 1:32 Scale Steel Model Tests
1984 – 1:8 Scale Steel Model Tests
1987 – 1:6 Scale Reinforced Concrete Model Test
1988 – F4 Phantom Jet Impact Test
1994 – Containment Bellows Test
2001 – 1:4 Scale Pre-stressed Concrete Model Structural Failure Test
2005 – OECD/NEA/CSNI ISP#48 on Containment Capacity

Fig. 6 Pre-stressed concrete containment failure test [SANDIA2006]

In 1988, Sandia conducted together with Muto Institute of Structural Mechanics, Japan, an impact test of a mockup of a containment. The purpose of the test was to determine the impact force, due to the impact of a complete F-4 Phantom — including both engines — onto a massive reinforced concrete target (3.66 m thick). The impact occurred at about 215 m/s (770 km/h), see *Fig. 6*. The test was not intended to demonstrate the performance (survivability) of any particular type of concrete structure to aircraft impact. The effects of fire following such a collision were not a part of the test. The test established that the major impact force was from the engines.

Fig. 6. Impact test at Sandia National Lab, USA performed 1988 [Sandia2006].

6.2.2 Europe and NUGENIA

Nugenia is an association of over 80 participants dedicated to R&D of nuclear power with focus on reactors of today. The stakeholders are mainly industry, research and safety organisations, committed to develop joint R&D projects. The collaboration builds on the past success of a European Commission-supported network called NULIFE, and integrates a working group from the Sustainable Nuclear Energy Technology Platform (SNETP) and SARNET. The work of NUGENIA is organised in several technical areas, such as Plant safety and risk assessment, Severe accidents, Integrity assessment of Systems, Structures and Components, Fuel Development, Waste and Spent Fuel Management and Decommissioning. Several projects have been performed in a European context, such as Longlife Perform60 (RPV and internals), ACCEPPT (Concrete) and ADVANCE (cable ageing). These were all focused on ageing studies. The present list of project proposals is long and will be shortened in connection to the selection at the Horizon2020 call in 2014.

VERCORS is a French program of experiments to be carried out by EDF at a new experimental 1/3 scaled RCB mock-up in Renardières near Paris from early 2015. The mock-up will be finely instrumented to monitor its behavior of the concrete, both on the rebars and on the prestressing cables, with parameters such as resistance, modulii, drying, shrinkage, creep, and permeability. The main objectives of the project are to study:
the behaviour at early age, the evolution of the leak tightness under aging (drying effects are about 9 times faster on the mock up because of scale effects), and behaviour under severe accident conditions.

7 Discussion and conclusion

7.1 Overall benefits for the present fleet

The present nuclear fleet is ageing, with an average age in the world of almost 30 years. Over 60 of all reactors in operation are 40 years or more. However, more than 70 units in the United States have been granted LTE of up to 60 years.
For the Nordic countries, the average reactor age is 35 years, with 4 reactors that have operated for more than 40 years in 2015. *Fig. 8* presents the number of operating reactors as function of age in the Nordic countries and in the world.

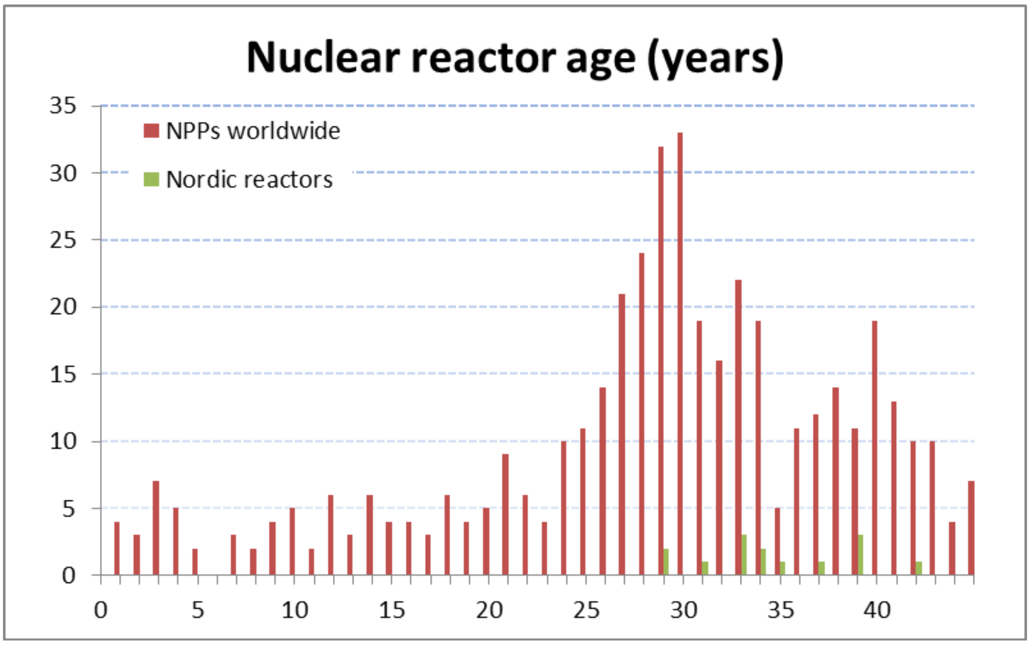

Fig. 7. Number of operating nuclear power plants (in total 435) as a function of reactor age based on data from IAEA PRIS data base [IAEA2014].

Due to the ageing fleet increased focus has been put on ageing and long-term operation (LTO) studies recently. With a shutdown complete site operated from the same generation of plants, with the same manufacturer and licence procedure as most of the Nordic BWRs, it is valuable to be able to study several aspects. Apart from the pure ageing phenomena that could be studied, there are several issues of interest also for the operation of the present fleet. One such thing is the already ongoing training programmes at Barsebäck, where highly skilled staff can improve maintenance procedures, and where less experienced staff can learn how a nuclear plant works from the inside without any risk of disturbing operation.

The benefit of onsite experiments and projects is highest for large non-replaceable components, mainly the reactor pressure vessel, and the reactor containment. Full-scale validation of codes is also of high priority in such programme.

7.2 Risks involved

The benefits of the discussed proposals in this report should be put in perspective of several risks involved. One risk is that the experiment is not representative enough, either of any specific plant, or to any specific test case considered. This should be able to verify in the planning phase well in advance of further consideration.

Several of the suggested projects are costly, time consuming and with high risk of altering dismantling plans, such as studies of fuel pool boiling, or fire studies. Special care is needed to justify such extreme studies.

7.3 Time plan

The short-term experiments are foreseen to run from 2015-2018. Mid- to long term projects will be carried out from 2017 to 2022 according to access of facility and overall decommissioning planning. See also *Fig. 9* below for the time line of Barsebäck and this R&D Arena put in perspective. It should be noted that time-shifts are possible in the plan due to the overall dismantling plans (such as decisions on interim storage on site with municipality acceptance in June 2014), delay due to training sessions, transport issues etc.

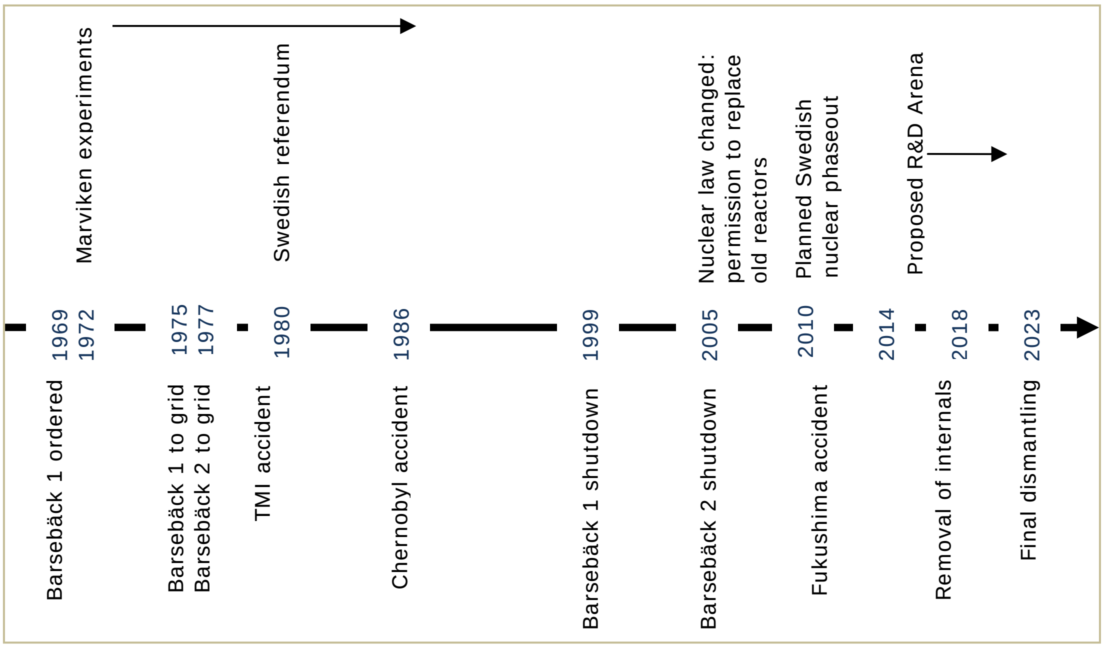

Fig. 8. Timeline of Barsebäck operations and related activities

7.4 Coordination and steering

An overall coordination of ageing studies needs to be defined. Several possibilities within already formed groups exist, but none with the full scope of such broad ageing study suggested here. It is therefore suggested to establish a programme committee, the R&D Arena Steering Board, consisting of Elforsk members. To start with, it is suggested that the Elforsk Nuclear Programme Board will act as the Steering Board for this R&D programme.

A list of suggested pilot projects, of which a few could be considered in a short-term, is given below:

- **Digital construction input (Mesh) of containment**
- **Preparation of tendon research programme proposal** for analysis of forces in the Barsebäck tendons
- **Status and condition of irradiated concrete**
- **Extraction of concrete cores to study the damaged aggregates** in the biological shield
- **Extraction of concrete cores to determine the moisture distribution** in the outer cylinder wall and verifying existing models of concrete drying.
- **Limited study on the identification of internal components** for material studies at B1 and B2, with clear needs and connections to the present fleet of operating reactors.
- **Mechanical testing of available surveillance specimens in Barsebäck 2** for validation and extension of a Nordic data base on RPV embrittlement.
- **RPV material (trepan-sample)** for studies on attenuation through the RPV wall.
- **Statistically significant evaluation of metallurgic variability in RPVs.**
- **Variability in the depth of the mixed zone between stainless steel and RPV wall.**
- **Material sample library and archive for future R&D needs**

Results from pilot projects are to be presented in a workshop together with international expertise. Discussions on further project proposals, funding and coordination procedure is envisaged.

7.5 Project proposal process

The projects proposed under the Barsebäck R&D Arena are to be presented and prepared in four steps:

1. Experimental proposal with objectives and technical needs is specified and discussed in the already established expert networks discussed in this report, such as Elforsk Concrete Research Programme, Materialgruppen, Beräkningsgruppen, NBSG et c. for a first review.
2. Funding and participants are suggested to the Barsebäck R&D Arena steering board together with recommendation from step 1.
3. The Elforsk Nuclear Programme Board acts as the Barsebäck R&D Arena steering board, and discusses the funding, time frame, dependency and synergy with other proposals. The board will give a recommendation to BKAB.
4. Barsebäck is deciding whether to give a go-ahead or not to the project.

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Appendix A: R&D Arena Barsebäck

Background

The operating time of the Swedish nuclear power plants will be increased from the original 40 years to at least 50 years. In order to insure a safe operation, ageing effects must be investigated through tests, modelling and verification. Destructive testing and investigations causing, or being associated with, risk for substantial unavailability can only be carried out to some extent in operating power plants and are more suitable for plants that are no longer in operation.

The Barsebäck reactors were transferred to service operation in 1999 and 2005, respectively. Awaiting the decommissioning, a business function has been started working with education, selling components, research etc. The reactors are though basically intact, and it is possible to use the facility for research and tests at least until 2021, as long as it avoids disturbing the training and education activities and the planned removal of the internals in 2016-2018.

Elforsk coordinates some Research and Development (R&D) for the Swedish utilities and nuclear regulator. The company is coordinating research activities within the field of nuclear in the areas of concrete structures, polymers, I&C and vibrations. There are also a number of collaboration groups outside Elforsk working with nuclear research financed by the Swedish nuclear power plant owners, for example within metallic materials, calculations, etc.

Workshop

E.ON in cooperation with Elforsk initiated a workshop on February 11th to discuss the possibilities of using Barsebäck as an R&D arena, taking samples and performing tests, before it is decommissioned. The workshop gathered some 15 representatives from the Swedish and Finnish nuclear power plants and from the owner companies of the plants. During the discussions three possible areas of research/testing were identified, see table A1 below. It was also clear that it is vital that the activities are coordinated in order to secure that the tests that are performed doesn't destroy the prerequisites for coming test.

Table A1. Potential areas for Barsebäck R&D Arena

Elforsk coordination		
Aging	Operation	Decommissioning
• Concrete • Metallical materials • Polymers • Etc.	• Flow • Fire • Tightness • Etc.	• Clearance • Work environment • Etc.

It was also stated that existing forums, for example the concrete research program and the cooperation group for metallic materials should be used in the planning of the research program to avoid forming parallel groups. During the discussions the focus was on the areas ageing and operation, since large resources are already put into the area of decommissioning in other forums. A teamwork was performed where the participants were divided into three groups discussing the need for research and testing within the areas metallic materials, concrete structures and operation.

Appendix B: Questionnaire

Elforsk has initiated a study of possible research areas to be carried out at the shutdown nuclear plant Barsebäck, and the potential of an R&D programme on this subject. I am at the moment collecting inputs to this study, and would like to have some inputs on your possible interest in such study. The background to this study is detailed in appendix A. The study should be finalized by June 25, and the research activities could probably start in January 2015. It should be noted that all projects are intended to be financed by the partners themselves within this programme.

A short questionnaire is attached below. I would appreciate all your help in collecting an as broad view as possible to the potentials of using Barsebäck as an R&D Arena.

Questionnaire and suggestions for activities in R&D Arena Barsebäck

Name: *Organisation:*

Research interest and focus area:

Previous studies at Barsebäck, or in a similar situation:

Do you see an interest for other stake holders (research groups, industry…) in this programme? Which groups?

Are you involved in, or do you see, international collaboration related to this?

Should other R&D areas be included in such programme? Which?

Proposals for activities in this programme:

Area and proposal **Name and short description** **(add below table if necessary)**	**Time frame and duration** **Short term: within 1-2 years** **Long term: 3-8 years**	**Complexity and cost** **(used to estimate cost and plant involvement)**
Example: Ageing. RPV specimen testing *Specimens irradiated during operation*	*Short term, 2 months planning*	*Simple: Existing specimens available. Development of tools* *External costs: transport, testing…*

Other comments to the overall scope:

Thank you for your cooperation!

Appendix C: Barsebäck design details

The Barsebäck plant.

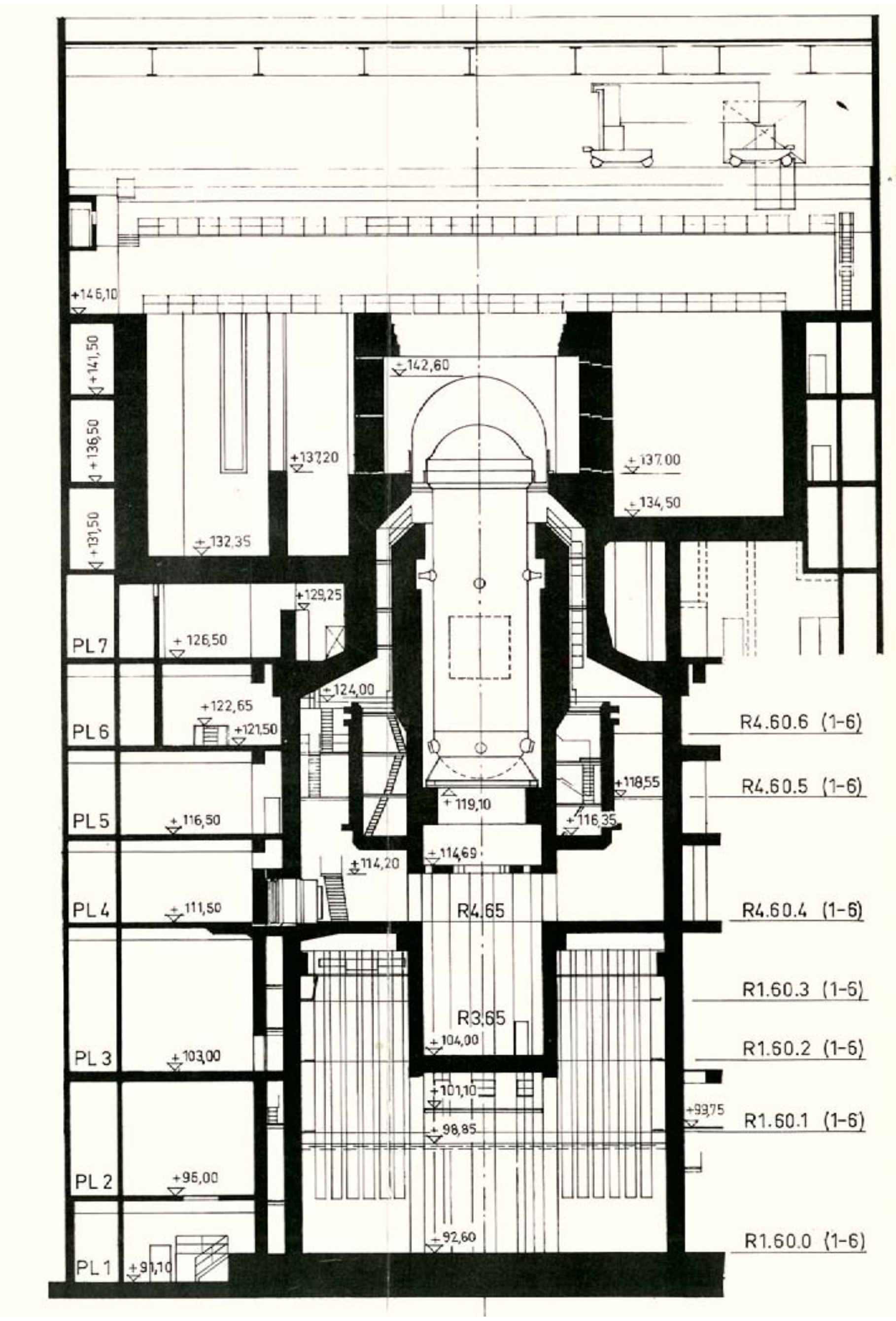


Fig. C1. Barsebäck vertical overview with reactor pressure vessel and containment building

Table C1. Basic facts about the Barsebäck 1 and 2 plants (similar values for both units).

General	
Thermal reactor power, MW	1800
Output power, MW	600
Number of circulation loops	4
Operational pressure, bar	70
Total coolant water flow, kg/s	7700
Steam flow, kg/s	900
Steam pressure, MPa	7,0
Steam temperature, °C	286
Feedwater temperature, °C	184
Reactor Pressure Vessel	
RPV weight, tonnes	530
RPV total height, m	20
RPVV inner diameter, m	5,45
RPV thickness, mm	126
Fuel	
Number of fuel elements	444
Fuel weight per element, kg UO_2	181/172
Number of fuel rods per element	100/96
Fuel cladding	Zr_2
Fuel pellet diameter, mm	8,19
Control rods	
Type	109 cross type blades
Absorber material	Hafnium, Boron carbide
Driver mechanism	Electro-hydraulic
Main circulation pumps	
Pump types	4 centrifugal pumps
Flow, m^3/s	2,55
Pumping capacity (height), m	55

Turbines	
1 high-pressure, three low-pressure	
Turbine power, MW	615
Efficiency, %	33,7
Steam flow, kg/s	813
Humidity level in primary steam, %	0,004
Steam pressure/temp	
Before high-pressure turbine, MPa/°C	6,1/284°C
After high-pressure turbine, MPa/°C	0,65/162°C
Before low-pressure turbine, MPa/°C	0,62/256°C
In kondensor, MPa/ °C	0,043/30°C
Outlet temperature, °C	256
Outlets and area per turbine, m²	6x8,5
Sewage blade length, cm	95
Rpm	3000
Flow capacity in condenser, m³/s	22,2
Coolant pumps per turbine	4
Dump capacity, %	100
Total length of turbine, m	50
Generators (one per unit) from ASEA	
Power, MW	620
Voltage, kV	17,5
Rated power, MVA	710
Cooling (stator/rotor)	Water
Transformer	
Rated voltage, kV	17,5/410
Rated power, MVA	700
Emergency power 4 diesel generators	
Manufacturer	MTV
Nominal power, kW	1600
Rated voltage, kV	6,3
Rated power, MVA	2,1
Rpm	1500

Appendix D: List of project proposals

Below follows a table of proposed experimental activities and proposals at Barsebäck. Conclusions from the Chalmers ageing report [Liljenzin2013] are also included below.

Experimental proposal	**Area**	**Material**
Boiling in fuel pool (leak-test and validation of models)	Safety	Concrete
Material samples (trepan) from RPV	Ageing	Metal
Test (surveillance) specimens from RPV	Ageing	Metal
Water hammer study (Loads due to pressure transients)	Safety	Metal / Water
Surface crack systematics	Ageing	Metal
Polymer validation and comparison B1 / B2	Ageing	Polymers
Polymer study on cable penetrations and sealings	Ageing	Polymers
Anchor plate study	Ageing	Polymers
Analysis of rotor blades in turbines	Ageing	Metal
Tubing in intermediate re-heater and valves	Ageing	Metal
Study of stellite / surface material in valves	Ageing	Metal
Stainless steel components in valve housing	Ageing	Metal
Stainless steel internal structural components	Ageing	Metal
Study on irradiated X750 alloy	Ageing	Metal
Weld between Ni-based plate and SS components	Ageing	Metal
Change in ageing of materials in different environments	Ageing	Polymers
Condition measurement of sensors and switches	Safety	Polymers
Surface protection paints on components and building	Ageing	Polymers
Ageing of lubricants in containment	Ageing	Polymers
Ageing of nickel based alloys (e.g. Alloy 600, Alloy 82, Alloy 182)	Ageing	Metal
Electric ground gridline degradation	Ageing	Metal (copper)
Cracks in Turbine rotor blades	Safety	Metal (SS)
Corrosion inspection in reheaters and tubings in heat exchangers	Ageing	Metal (SS)
Tendons: long-term testing (tensile force and relaxation, corrosion)	Ageing	Metal/Concrete
Component status and surface structure on valves	Safety	Metal
Material structure through RPV thickness for attenuation study	Ageing	Metal
Ni-based allow welds onto SS in core spray system	Ageing	Metal (SS)
Slack reinforcement condition, cable conduit	Ageing	Metal/Concrete
Concrete in outer cylinder wall, crack systematics	Ageing	Concrete
Concrete liner (cladding) condition	Ageing	Metal/Concrete
Polymer material in concrete	Ageing	Polymer/Concrete
Air ingress in concrete: containment over-pressure	Safety	Concrete

Experimental proposal	Area	Material
Cooling water-way structure	Safety	Concrete
Irradiated concrete and containment specimen testing	Ageing	Concrete
Flow rate in chimney and smoke diffusion in case of fire	Safety	Fire/gas
Flooding, loss of coolant and pipe breaks	Safety	Water/Steam
Test of FILTRA fresh air filtering system	Safety	Water/Steam
Measurements in the biological shield, analysis of humidity	Safety	Concrete
Microscopic study of concrete cores from biological shield	Safety	Concrete
Humidity distribution in outer wall concrete core to study drying	Safety	Concrete
Dynamics in pools, stratification, and exchange dry- and wet well	Safety	Water
Containment liner deformations and strains during pressure test	Safety	Concrete
Wet-well blow-down and induced vibration in critical locations	Safety	Concrete
Test reactor building with impact load, induced vibrations	Safety	Concrete
Benchmarking impact load and induced vibrations	Safety	Concrete
Depressurisation of containment	Safety	Concrete
Fire studies in confined areas: relay station, turbine, battery backup	Safety	Fire/gas

Proposals included from the conclusions in the ageing report entitled "Ageing of components in nuclear power plants" [Liljentzin2013] (note that the priorities are based on personal views by the authors):

Priority	Proposal
Basic Material Research	
Low	Embrittlement
Low	Fatigue
Polymer ageing	
High	Sealing components
Medium	Circuit boards
Low	Cables and insulation
Ageing of mechanical components	
Low	Abrasion and wear
Low	Corrosion
Low	Fatigue
IC- and Control-systems ageing	
High	Relay ageing; Nowadays two backups, historically only one
High	Software ageing and drift
Medium	Hardware ageing
Medium	Old circuit boards: Sn and Cu growing

Priority	Proposal
Concrete ageing	
High	Compilation of material property data for long-term performance and trending
High	Effects of low irradiation field and long-time exposure of concrete
High	Effects of the very slow concrete flow dimensions and stability
High	Non-intrusive methods for inspection of thick-walled structures
High	Utilization of structural reliability theory incorporating uncertainties
Medium	Application and performance (e.g., durability) of repair materials and techniques
Medium	Assessment and validation of non-destructive evaluation methods
Medium	Dry out (“burn off”) velocity of the concrete
Medium	Global inspection methods for metallic pressure boundary (steel liner)
Medium	Improved damage models and acceptance criteria
Medium	Knowledge about ageing from bridge building and military application
Medium	Probabilistic modelling of component performance
Low	Evaluation of environmental effects on concrete
Low	Improved constitutive models and analytical methods

Appendix E: Details on some proposals

Below follows brief descriptions of some the high- and medium level proposals listed in Section 7 and Appendix D above. The numbering refers to Civil Works (C), Metallic materials (M) and Other Examples (E).

C.1 Digital construction input (Mesh) of containment

This project aims at developing and creating the Mesh of the reactor containment, for further use in numerical analyses. Nowadays, calculations and modelling of concrete structures and buildings are carried out by numerical methods. The construction geometrical and material data are stored in digital blue prints to be used for meshing the construction. Every mesh is given coordinates (in 3D) and material specification. This mesh is – together with other documentation – an important prerequisite for correct description of the construction. This mesh can further be imported to different types of computer codes.

A digital mesh of Barsebäck would be made available for project groups that plan to be actively performing projects within the Barsebäck R&D Arena which will facilitate the effective use of results, and information transfer between projects, a possible way to well evaluate the research work, and the requirement to transfer results from Barsebäck to other reactors.

C.2 Analysis of Barsebäck tendons and proposal on research programme

The goal of this project is to specify an integrated research programme for the determination of tensile forces in the Barsebäck tendons. Included is:

i. a basic description of the theoretical and experimental analysis methods that should be used for correct determination of forces in the tendons,

ii. a verification of theoretical models,

iii. possibilities to transfer results from Barsebäck to other nuclear power plants.

The project will result in proposals on measurement and surveillance methods, and to take into account the work that is ongoing at other plants. The study is going to result in a plan for further work on the evaluation of the tendon status in Barsebäck. The results are going to include proposals on sub-projects that can be performed within Barsebäck R&D Arena.

The research programme is intended to be used for

i. organisation and management of R&D related to tendons

ii. proposed sub-projects and choice of project responsibles

iii. management of sample removal and execution of tests

iv. project management and synergies to other ongoing Barsebäck projects

v. information exchange with other nuclear power plants and coordination of their work.

After the execution, the results will contribute to an increased knowledge base in the area of tendons, and to improved and validated computational models. These results are of importance for all plants owners, both where tendons and rebars are used, where it will improve the overall understanding.

C.3 Status and condition of irradiated concrete

This project will study the irradiation-induced effects on concrete properties, based on irradiated construction parts from the reactor containment in Barsebäck. This project need to be coordinated with the work already carried out at OKG, where an analysis of concrete cores is underway.

Results from the project will show whether concrete in a radiation environment, such as the one in the Barsebäck containment, is damaged or not. The degradation mechanisms are to be clarified, and a general model for a life-time estimate will be developed.

C.4 Extracting concrete cores for study of damaged aggregates over the depth of the structure

When exposing concrete for radiation this may result in a degradation of the material. The main degradation occurs in the aggregates which swell and crack due to the neutron radiation. The hypothesis for this study is that it is possible to measure at what depth the radiation has affected the concrete, by studying crack formations in the aggregates. This is done by extracting concrete samples from the biological shield and with microscope view the eventual crack formations in the aggregates. By performing this gradually, from the inner surface to the outer, it shall be possible to determine to what depth the concrete has been affected. The aim is to use a simple microscope with a magnifying capacity of X20 to X40 and photograph the aggregates with a digital camera. These observations can in addition be complemented with mechanical testing of samples from different depth, in order to see if there is a correlation between the affected concrete and the mechanical properties. The findings form the tests can in a third part be used to compare with results from international studies in the area off shielding properties of concrete. Knowledge about what depth the concrete has been affected on can help in the work with decommissioning regarding clearance of concrete components. It is also of interest for long term operation if the concrete has degraded.

The preparation phase includes a state of the art study and an evaluation how the microscope study has to be performed. The preparation phase also includes development of the equipment needed.

The Execution phase includes the extraction of the samples. The extraction is done at site using a core drill. Two to three cores is extracted from the biological shield from different zones, both in areas with high radiation and areas with lower. The core is split into two halves using a concrete saw.

The analysis phase includes the material testing, the microscope classification and to write a report.

The expected zone that has been affected by the radiation is only a couple of dm, hence it should be favourable to have access from the inside of the biological shield. In order to achieve this, the RPV has to be removed prior to the core extraction. As an alternative the core can be taken from the outside of the biological shield, this is however a more extensive work due to the scale of the structure. The study should not start before Q1 2016. The extraction of the core samples should not affect the operation and not affect other tests (Such as LOCA tests) either. In order to perform the mechanical analysis the concrete samples has to be taken to a laboratory. If the concrete is contaminated this will not be possible. In that case this part of the study has to be excluded. Regarding the microscopic analysis the aim is to do all the work on site. This is done by using a small and portable microscope that enables to take high quality digital photos that can be analysed at another location.

C.5 Extracting concrete cores to determine the moisture distribution

The moisture conditions and distribution in the containment wall at Barsebäck has been studied in an earlier PhD project performed by the Division of Building Materials at Lund University. In that study the predicted moisture gradient in the structure was assessed using a model and moisture measurements from the surface of the outer cylinder wall. In contrast to the PWR:s the temperature gradient in a BWR is significantly smaller then in a PWR, one question is thereby if the temperature gradient is so large that the hygro-thermal moisture transport has to be accounted for or if a isothermal condition gives a good enough estimation.

The measurements are performed using the same monitoring equipment used in the PhD project, climatic conditions inside nuclear reactor containment. The equipment is installed in drilled holes and left for at least one week or more. The results are thereafter compared with results presented in previous studies and with new results from new models describing the moisture transport. One problem is that the containment has been exposed for a “new” climatic condition for the last 10 years which has to be taken into account when comparing the results.

A state of the art on the field is conducted in order to evaluate which models shall be used in the study. During the preparation phase the equipment used has to be assembled and checked. A plan for the monitoring has to be established.

During the execution phase the equipment is installed at site and demounted after about one week. The work requires two entrepreneurs to assist with drilling. The installation and decommission will take one day each - on site.

In the analysis phase, the results will be compared with results from calculations. The work will be done in association with e.g. LTH. The study does not affect the operation or other studies, e.g. LOCA tests.

When drilling the holes for the installation of the equipment, the reinforcement has to be avoided. This can be an obstacle at larger depths and may result in extra on-site work. It is also very important to avoid the pre-stressed tendons. Extra attention has to be set on these.

M.1 Investigation of most important components for further study

This proposal contains a pre-study includes a detailed discussion with Barsebäck on available components of stainless steel (SS) internals. Links to previous studies.

The execution phase includes the complete planning of detailed component removal, transport, mechanical testing, TEM/SEM/ Atom Probe methodology. Prepare proposal for PhD study on the subject. The study could start in 2015, in close collaboration with Materialgruppen.

M.2 Material testing and analysis of irradiated surveillance specimens

This proposal is straight-forward in the use of already irradiated materials from the surveillance capsules in Barsebäck 2, for direct transport to a suitable lab for further material impact and tensile testing. There are three complete sets of specimens available in Barsebäck, and a possible collaboration is already established between FKA, RAB, OKG and TVO on this issue.

The specimens represent neutron fluences similar to 40 years of operation in BWRs as well as one set of accelerated ageing corresponding to doses well beyond the lifetime of PWRs. This would be valuable in solving uncertainties how the Nordic (Uddcomb-manufactured) RPV welds behave for LTO and beyond. It would be beneficial both for BWRs and PWRs, including a comparative analysis on of low and highly accelerated ageing.

M.3 RPV material from trepan samples

Best practice investigation on cutting techniques and experience from previous studies. During the preparation phase the equipment used has to be assembled and checked. A plan for transportation, mechanical testing, microscopy and safe storage has to be established.

An execution phase with trepan removal (two envisaged) includes to get equipment in place at site and prepare for removal during two weeks. Material samples are to be prepared by cutting into specimens for material testing.

The analysis phase includes a comparison of results with previous studies from surveillance specimens. The work will be done in association with academia (Chalmers or KTH foreseen).
It is suggested to develop a proposal for possible Post-Doc at Chalmers or KTH preferably. The study could start in 2015.

M. 6 Material sample library and archive for future R&D needs

Suggestion to prepare representative samples of a variety of materials at different positions for future use in research and material investigations. This pilot study aims at identifying suitable material pieces, and investigating possibilities for storage. A detailed mapping and documentation of samples would be needed.

E. 1. Containment pressure transients

Reactor containments consist of a steel liner embedded which acts as a leak-tight barrier. Transient air pressure distribution in concrete can be tested order to see verify/validate leakage rates obtained from simulations of transient air pressurization of concrete pores inside the steel liner as discussed for example in Ref. [Elforsk2009]. A model describing transient air pressure distribution in concrete pores was developed in the computer software Comsol Multiphysics, and a simplified finite differences model was also developed in Excel.

The permeability and other material properties of relevant concretes is important to monitor. This has been part of a previous study including a laboratory test setup used for verification. At two occasions measurements were done during a pressure test at Ringhals R2 and R4 respectively. Humidity and pressure profiles were measured at various points in the concrete structures. It is quite clear from the measurements that there is a tremendous rise in air pressure also at large depths, even after the pressure is lowered after the test. The simulation models require knowledge on the geometry and material properties of the concrete and the epoxy coatings at the concrete surfaces. The models could be improved also by including the effect of different relative humidity at different depths, the pressure dependency of the permeability and the dissolution of air into humidity in the porous system [Elforsk2009].

The function of the biological shield is important to maintain to limit risks of radiation. This is primary assured by water in the concrete regarding neutron radiation. Due to this, the humidity level in the concrete is important. This also links to the condition of the concrete structure due to possible degradation from 40 years and more of operation, e.g. radiation and embrittlement [Oxfall].

E. 2. Electric ground gridline degradation

The electric ground gridline at the nuclear power plants consists of inner and outer systems of copper wires in a grid pattern mainly buried in the ground why it is difficult to verify the status. The grid pattern covers a large part of the plant area with extension into the water inlet channels.

At Barsebäck, these copper connections could be verified during decommissioning to see possible degradation, both regarding individual wires, and joints between different sections. The verification and analysis of these wires and different joints, both on land and in the water inlet, would be beneficial in estimating the condition at the operating plants.

E. 3. Pressure transient (water hammer) study

Pressure transients e.g. water hammer, seldom causes pipe system- and component failure, even though such examples exist. When the plants were constructed the topic was hot and many problems were identified and components were designed and optimised world-wide through research programmes. However, the main issue is that the analysis tool RELAP5 is sometimes limited in the accuracy of models of the components on one hand and the physics on the other hand. Such limitations often lead to the necessity to use over-conservative assumptions, which in turn lead to system pressures and forces exceeding the limits.

Appendix F: Marviken experiments

Below follows a summary of reactor safety experiments carried out in the Marviken nuclear power plant close to Norrköping in Sweden between 1972 and 1986 [Thorén]. The experiments covered mainly international collaboration on containment experiments with simulated tube breaks, with the objective to demonstrate the pressure-suppression principle and to verify calculational tools and models. The participants included

- Forsöganlaeg Risö, Denmark,
- Institutt for Atomenergi, Norway,
- CEA, France,
- GEsellschaft für Kernenergiververtung in Schiffbau und Schiffart (GKSS), Germany,
- JAERI, Japan,
- KEMA, The Netherlands,
- US NRC, USA.

Three experiments were performed between 1972 and 1978 on full-scale study of critical flow during tube breaks. In some of the experiments at Marviken focus was also on how fission products behave during tube breaks. This was later complemented by studies on aerosols leaking out of the reactor core cooling system. This continued until 1986 when all the experiments of Marviken ended. The development of FILTRA by Asea Atom could also be tested and later on installed in all BWRs by Asea Atom in a system called Multi Venturi Scrubber System.

Below follows a description of one of these experiements, as noted in one of the OECD-NEA reports on Marviken experiments [NEA2000].

Title: **Marviken: Pressure Suppression Containment-Blowdown Experiment No.18.**

Objectives: Obtain insight into modeling capabilities of containment computer codes with respect to

- vent clearing transient
- steam/air transport through vent system
- medium-time pressure suppression characteristics
- periods of oscillatory condensation and/or chugging not considered

Facility: Containment with pressure suppression system of the decommissioned MARVIKEN nuclear power plant. Containment involves multi-compartment drywell part (1,970 m^3) connected to the wet well (2,149 m^3, 556 m^3 water pool) by channel-type vent system and header 58 vent pipes, 5.5 m long, 0.3 m diameter; 30 vent pipes blocked during the experiment. Total vent flow area activated during the experiment 1.95 m^2, initial submergence of vent pipes 2.8 m.

Scaling Information: Commercial containment, however no similarities to BWR typical pressure suppression system containments

Parameters offered for Comparison:

- 5 absolute pressures
- 5 local pressure differences (e.g., drywell-wetwell)
- 5 gas phase temperatures
- 1 water pool temperature (averaged)

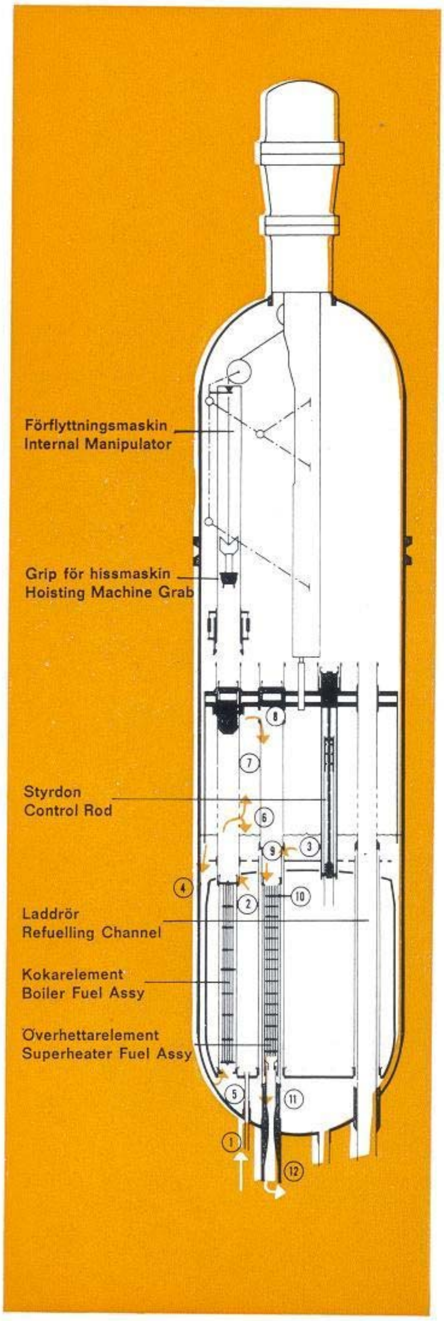
Dominating Experimental Uncertainties: nothing mentioned in the report

Findings:

- relative to the accuracy of the measurements, the calculated pressure and pressure differences between drywell and wetwell agreed well with experimental data
- calculated time to clear vents slightly too short
- heat transfer between fluid and structures in the drywell underestimated due to low heat transfer
- coefficients
- small number of nodes sufficient to meet experimental drywell results
- important issues not addressed by this ISP (e.g. periods of oscillatory condensation and/or chugging)

Recommendations: nothing mentioned in the report
Total Duration of Exercise: June 1983-August 1984

Participation: 4 institutions from 4 countries
Codes involved: CONTEMPT-LT, ARIANNA, ZOCO-V, COPTA



FLÖDET I REAKTORTANKEN

FLOW THROUGH REACTOR VESSEL

Flöde Flow	Temp °C	Mängd Mass kg/s
1. Matarvatteninlopp i tanken Feed water inlet to vessel	109	204
2. Vatten ut ur moderatorn Water out from moderator	224	204
3. Vatten in i överhettarnas isolerrör Water to superheater isolating tube	260	150
4. Vatten ned genom fallspalten Water down through downcomer zone	260	1800
5. Vatten in i kokarelementen Water to boiler elements	260	1950
6. Ång/vatten-blandning ut ur kokarelementen Steam/water mixture out from boiler elements	263	1950
7. Vattenavskiljning Water separation	263	216
8. Ånga in i överhettarkanalerna Steam into superheater channels	263	215
9. Ånga ned genom överhettarelementen*) Steam down through superheater elements	263	215
10. Ånga i förbindelse med isoleringsspalt Steam in connection with insulating annulus		litet small
11. Överhettad ånga ut gm överhettarnas utloppsdysor Superheated steam out through superheater outlet nozzles	475	215
12. Överhettad ånga till ångsamlingslåda & turbin Superheated steam to steam header & turbine	472	215

*) Vid drift med mättad ånga är denna kanal tom.
These channels are empty during saturated steam operation.

Fig. 9. Overview of flow paths through the Marviken RPV [Thorén2012].

Appendix G: List of contacts

Concrete
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General issues:
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