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PLANT LIFE MANAGEMENT AT LOVIISA

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SUMMARY

In the original design, the vendor guaranteed a 30-year lifetime for the plant. Separate lifetimes that could be higher were given to the main components. Generally, there is some conservatism in the assumptions and calculations behind these figures. New predictions based on the operating experience and more accurate calculations give a longer and more accurate lifetime prediction with less conservatism regarding fatigue.

In the original design some deterioration mechanisms, like thermal mixing in T-junctions and stratification in horizontal pipes, were not taken into consideration, because their existence was not realized at the time when the plant concept was designed.

Plant life management is one of the key processes in nuclear power plant operation. Loviisa NPP has developed a systematic management program for the plant life management. This development plan includes redefining of responsibilities and goals, extension and improvement of the condition monitoring programs and combining aging evaluations with long-term planning activities. The goal is that all maintenance and inspection activities should support plant life management.

The most critical components for the Loviisa plant's lifetime are the steam generators. They are very difficult to replace, due to their location deep inside the containment, surrounded by thick concrete structures lacking wide enough openings. Nowadays all steam generators are in relatively good condition, and only two tubes have been plugged so far.

Possible replacement of the reactor pressure vessel (RPV) is considered to be difficult but not impossible. However, its lifetime can be extended by thermal annealing, because the main deterioration mechanism is radiation embrittlement. The critical circumferential core weld of Loviisa 1 RPV was successfully annealed during the refueling and maintenance outage in August 1996.

Other components that may become critical in the long term are the main coolant pumps. The pumps are the only ones of their kind, and the original supplier has lost much of the know-how, due to the long timespan. It would also be very difficult to replace the pumps with other ones delivered by some other supplier.

In its operating philosophy, unnecessarily heavy thermal loads are avoided, and the plant is being operated smoothly.

Among other things, a lot of pressure transmitters, containment wall instrument penetrations and some electrical cables have been replaced.

Erosion-corrosion is kept under control by an extensive program including wall thickness measurements, weak point analyses performed by a computer code designed for erosion-corrosion and replacement of pipe sections with a more erosion-corrosion resistant material. Also the implemented change in the secondary side water chemistry is expected to decrease erosion-corrosion.

The screening of thermally loaded locations and loading mechanisms is under way. Attention is paid to leaking valves and pipe sections with missing insulation. A new on-line monitoring system will be installed in the next refueling outage to monitor temperatures at the locations with presumable thermal stratification.

INTRODUCTION

Imatran Voima Oy (IVO) has in Loviisa on the southern coast of Finland a nuclear power plant which consists of two VVER-440 units. Their commercial operation began in 1977 and 1980 respectively. The nominal gross electrical power per unit was originally 440 MW, but, due to the favorable cooling water conditions and a comprehensive modernization program, this has been increased up to 510 MW. The units operate excellently. In recent years, the load factors have mostly exceeded 90 %, see attachment. The plant design is based on the Soviet concept, but has been adapted to Finnish conditions and safety requirements. The designed life of the components is 30 years, in general, but 40 years for the reactor pressure vessel. The long-term goal for the critical components is 50 years of operation.

IVO has developed a company-wide approach to plant life management. The first stage of plant life management comprises the operational and maintenance histories, design and plant inspection data, using advanced computer systems. The life of the plant can then be controlled by maintenance, refurbishment & inspection programs, and by varying the method of plant operation. On-line monitoring is needed, and cost control and training must be taken into account if the life of the plant is to be managed efficiently.

Identifying the life-limiting factors is essential at Loviisa. In the following it has been concentrated on the aging taking the form of degradation of materials because of fatigue, erosion, corrosion, radiation and thermal effects. Certain other life-limiting factors will also be mentioned.

TYPICAL FEATURES OF THE LOVIISA PLANT

In the original design, the vendor guaranteed a 30-year lifetime for the plant. Separate lifetimes, that could be higher, were given to the main components. Generally, there is some conservatism in the assumptions and calculations behind these figures. New predictions based on the operating experience and more accurate calculations give a longer and more accurate lifetime prediction with less conservatism regarding fatigue.

An NPP operating license is issued in Finland for a limited time, and it can be renewed only after a thorough safety review which also includes the assessment of ageing. The current 10-year operating license of both Loviisa units will expire at the end of 2007.

When developing the lifetime management policy for the Loviisa plant, it has been emphasized that it must incorporate relevant operation and maintenance activities, and not be a separate issue. Support to lifetime management is provided by continuous improvement of the inspection and monitoring methods and by an extensive research program which studies ageing in surveillance samples representing pressure bearing materials, I&C parts, cables and concrete structures.

Plant life management is one of the key processes in nuclear power plant operation. Loviisa NPP has developed a systematic management program for the plant life management. This development plan includes redefining of responsibilities and goals, extension and improvement of the condition monitoring programs and combining ageing evaluations with long term planning activities. The goal is that all maintenance and inspection activities should support plant life management.

In operation, the plant management especially stresses the importance of gentle treatment of all equipment during normal operating states and tests, i.e. limiting, as much as possible, the number and magnitude of thermal and hydraulic transient loads and maintaining the water chemistry parameters in all cooling systems within their specified limits. Due to the different operating practices, it can be expected that the aging of Loviisa main equipment proceeds more slowly than at most other VVER-440 plants.

The operating and test procedures have been developed actively by the entire operating organization with the aim to reduce equipment loads. A good tool supporting the development is the process computer.

In the water chemistry area, not only the primary coolant is under careful control but also the secondary circuit water quality has been under close control throughout the plant operating life.

The maintenance strategy is based on:

- assessing the equipment condition from reliability records and from results of in-service inspections, tests, and condition monitoring
- active bilateral exchange of information with other VVER-440 plants, and on incorporation of lessons learned from other plants into the test and maintenance programs
- redefining service and overhaul intervals as need arises from the condition assessment
- repairing or replacing the equipment before they fail

A more detailed overview on VVER-440 plant aging is given e.g. by Laaksonen (1997).

RECENT AND FUTURE ACTIVITIES WITH LIFE-EXTENSION AT LOVIISA NPP

The plant has recently started a project that has three goals:

1. To re-evaluate the original strength calculations. The aim is to eliminate extra safety margins. This can be done by making the assumptions more precise and, especially, by

comparing the actual usage history with the calculation assumptions. Typically, the number of load cycles occurred is now, after about 20 years of operation, much less than what has been assumed in the strength calculations. Also, most of the transients are less severe than assumed. So far it seems that only the number of leak tightness tests is approaching the original allowed number.

2. In the original design, thermal stratification and mixing phenomena were not taken into consideration. The systems are now under review, and the goal is to find and list the potential areas of thermal stratification and mixing. The stratification typically occurs in horizontal pipe sections with a closed or leaking valve. T-junctions with different flow temperatures are typical mixing areas.

3. A system will be developed to assist life management. It will be organized systematically and include, for instance, drawings from the critical areas, thickness measurement results and references to corresponding fatigue analyses.

LIFE-LIMITING FACTORS AND THEIR MANAGEMENT

Embrittlement and annealing of the reactor pressure vessel

The design life of the reactor pressure vessel (RPV) is, as mentioned, 40 years. Originally, it was assumed that embrittlement of the core region material would not cause any problems, despite the high neutron flux on the RPV wall. Surveillance test results at Loviisa, however, indicated that the embrittlement rate of the RPV steel was almost three times that of what had been expected. This problem was identified as the most critical aging phenomenon at Loviisa. To tackle the problem, some backfitting action was taken in 1980.

Several plant modifications and other measures have been carried out by the utility in order to mitigate embrittlement and to ensure integrity of the RPV. In spite of the applied mitigation measures, it was realized that the design lifetime of Loviisa 1 RPV could not be reached with adequate safety level without first reducing the embrittlement of the RPV materials. So far the only known method to recover radiation damage is thermal annealing. A post-irradiation heating of the steel to temperatures exceeding the irradiation temperature is known to make most of the irradiation-induced defects unstable and thus to recover mechanical properties affected by neutron irradiation.

It is obvious that this problem is generic to all VVER-440 plants. Consequently, the same kind of backfitting was later carried out at several plants of similar design. Certain older VVER-440 plants have, however, been operating with full core for so many years that the owners have been forced to anneal the RPV to recover the toughness properties of the core region weld.

According to the safety analysis, the highest acceptable transition temperature of the RPV is 140 °C. The material investigations showed that the transition temperature of the critical weld joint in the core area of Loviisa 1 RPV would reach 135 °C in 1996. In order to maintain sufficient safety margins and to respond to the requirements presented by the authority (STUK), the utility decided to anneal the core weld of Loviisa 1 RPV in 1996. Preparation and licensing work for the annealing started at the beginning of 1994.

The critical circumferential core weld of Loviisa 1 RPV was successfully annealed during the refueling and maintenance outage in August 1996. The weld was maintained for 100 hours in the 475 °C annealing temperature. The work was performed by *Škoda Nuclear Machinery* as the main supplier representing a consortium consisting of itself and *Bohunice Nuclear Power Plant* from the Slovak Republic.

A successful annealing was ensured by comprehensive thermal and stress calculations and monitoring. Pre-calculation work was independently performed by both *IVO Power Engineering* and *Škoda Nuclear Machinery* people. Temperature distribution during the annealing was followed by on-line monitoring. Also on-line stress field calculations were performed.

Comprehensive material testing programs have been carried out to ensure the licensing of the annealing. Part of these programs have not yet been finished, and the work is still going on. In the domestic program, a sophisticated reconstitution technique was used. Thus already tested halves of the surveillance specimen could be used as authentic material. The licensing work has mainly been carried out by the *Technical Research Centre of Finland (VTT)* and *Moht Otjig RM* in Russia. A new comprehensive surveillance program has been started to follow the re-embrittlement of the RPV after the annealing.

A more detailed overview on annealing is given e.g. by Hytönen et al. (1997).

Steam generators

The most critical components for the Loviisa plant lifetime are the steam generators. They are very difficult to replace due to their location deep inside the containment, surrounded by thick concrete structures lacking wide enough openings. Nowadays all steam generators are in relatively good condition, and only two tubes have been plugged so far. One tube was plugged due to a fabrication fault and another one due to wall thinning.

The original feed water manifolds are strongly eroded. Their material is carbon steel. It is not possible to replace them with new ones of similar kind, because they are located deep inside the steam generator between the tube banks. As a remedy, a new manifold construction has been developed to be placed above the tube banks. The material of the new construction is erosion resistant austenitic steel. The useless old manifold construction will not be removed, but left in its original position.

The cover plates of the primary collectors of a VVER-440 plant steam generator are heavily stressed and a generic problem. A cover renovating project is going on at the Loviisa plant.

Main coolant pumps and main gate valves

Other components that may become critical in the long term are the main coolant pumps. The pumps are the only ones of their kind, and their original supplier has lost much of the pump know-how, due to the long timespan. It would also be very difficult to replace the pumps with other ones delivered by some other supplier.

In the beginning of the plant operation, some cracks were detected in the covers of the pumps. The cracking mechanism has probably been environmentally assisted corrosion. The stress level in the cracked area was calculated to be about 400 MPa, which can be considered very high. The cracks were located in an area where cooling water from the pump was able to mix with the primary coolant. It was assumed that thermal fatigue has assisted in the crack formation. This problem was solved by improving the construction.

In the 1990s, there have been problems with the suction stools of the pumps. The suction stool is made of cast austenitic stainless steel. The part of it which leads the flow to the impeller is connected by eight round bars to the pump body. In these bars, there have been very deep cracks, in some cases, the bars have even been totally damaged. The reason has probably been flow-induced vibration assisted by thermal or corrosion fatigue. However, the vibration behavior of the fluid and the structure is very complicated, and the mechanism is not yet thoroughly understood.

In the beginning of the '90s, there were problems with the main gate valves. Several cracks were found in most of the main gate valves. The cracks were located in the stress concentration areas of the sealing faces, flanges and the welds. Small amounts of primary water had been able to leak into this crevice through the drainage lines. As a consequence, boric acid concentrated on the sealing faces, ultimately leading to stress corrosion cracking. The valves were repaired by machining and surface welding the sealing faces. To prevent the problem from re-appearing, the construction of the drainage lines was also changed.

Life management based on real-time analysis of measured data

In the original design, some deterioration mechanisms, like thermal mixing in T-junctions and stratification in horizontal pipes, were not taken into consideration, because their existence was not realized at the time when the plant concept was designed. To control these deterioration mechanisms and to monitor some other critical areas, an on-line monitoring system has been installed at the Loviisa NPP as part of the life management program (Rajamäki 1990). The system monitors temperature, strain and pressure data from the most critical components, and performs real-time fatigue analyses. The temperature and strain gauge measurements at the Loviisa plant have confirmed the presence of numerous fatigue loading cycles, which were not considered at the design stage. These fatigue calculations are based on an extensive finite element method (FEM) analysis, in conjunction with experimental monitoring results from the power plant.

After the preliminary measurements, thermocouples and strain gauges were installed permanently at Loviisa Unit 1 in the 1990 refueling outage in the following locations:

- pressurizer surge line nozzle
- pressurizer connection pipeline to primary circuit
- feed water nozzles of two steam generators
- the injection nozzle of the pressurizer

Water temperature fluctuations of the *pressurizer surge line nozzle* are monitored continuously. This measured history is used as input data for the crack growth calculations of a postulated crack in the dissimilar weld of the nozzle. The postulated initial crack size was selected considering the ability to observe any crack growth in the in-service inspections.

Strain gauges and thermocouples will be used to record the fatigue stress fluctuations of the *pressurizer connection pipeline* to the primary circuit. The main reason for the fatigue cycles in this pipeline is the thermal stratification of water. These cycles are used to calculate fatigue damage in the T-connection.

The fatigue crack growth of a postulated initial crack is calculated in *the feed water nozzle* of the steam generators. The hypothetical crack is situated in the weld root of the inner thermal sleeve. The measured water temperature history is the input data for these calculations. Strain gauges are also installed in the pipe elbow close to this nozzle to measure stresses due to water stratification in the feed water line.

Thermocouples are used to monitor the thermal fatigue loading of the *injection nozzle of the pressurizer*. The rapid temperature fluctuations during the injection are the input data for the calculations. Stress distribution and peak stresses in a notch inside the nozzle are calculated and used in the cumulative damage calculations.

The results obtained using this system show that the actual number of fatigue cycles is much higher than what was assumed in the design calculations. The calculated crack growth and the cumulative usage factor are, however, within the acceptance limits. The number of fatigue cycles with a low stress range, based on strain gauge measurements, is much higher than the number of similar cycles, based on the temperature measurements. In cases of slow cycles and cycles with a high magnitude, both of these methods give consistent results.

The monitoring system is going to be modernized, and the thermocouples re-organized in the refueling outage of 1998. The data collection system will be completely replaced. The measurements will be mainly focused to such locations where the temperature stratification is expected to occur. The locations are typically horizontal pipe sections with very low velocities or with no flow at all, or locations close to T-junctions with a remarkable temperature difference. Some measurement locations will be the same as now, and some others will be totally different. There will be two types of measurements: one consisting of 7 individual sensors and another consisting of 2 sensors.

After the modernization, there will be thermocouples installed in the following locations:
- pressurizer lower nozzle, surge line and branches to primary loops
- pressurizer upper nozzle
- feed water nozzles of two steam generators
- the emergency coolant pipes
- several locations in the coolant purification system
- emergency sealing water line
- the steam generators' blow-down line

Defects in the cladding of Unit 2's RPV

Even before Loviisa Unit 2 was commissioned, it was identified that the quality of the RPV cladding did not meet the design specification. During the final inspection of the RPV at the Izorsky factories in Leningrad, then USSR, it was found that there were numerous slag inclusions between the weld beads in the cladding. The largest inclusions were repaired by welding before the RPV was commissioned. Smaller inclusions which, according to crack growth calculations, would not grow to surface cracks during the RPV's lifetime, were accepted without repair. At the same time, an extensive co-operation program was started with research institutes in the former USSR to determine the behavior of the cladding, and, especially, the behavior of cracks in the cladding. The results of these studies have been published by Saario et al. (1989).

The main findings of the research program confirmed that the material parameters used in the crack growth analyses were conservative. In the in-service inspections of the RPV of Loviisa Unit 2, considerable emphasis was placed on the integrity of the cladding, the slag inclusions and the region under the cladding. A commercial NDT-service was used for these inspections. So far no crack growth or changes in the indications have been observed.

Erosion-corrosion of the secondary circuit

In the early '80s, it became obvious that erosion-corrosion of the secondary system piping and components would be an important factor to the safety and usability of the plant. Special features of the secondary circuit, such as neutral water chemistry and use of carbon steel, have made secondary system components and pipes susceptible to erosion-corrosion. The under-lying philosophy in the neutral water chemistry was to protect the most important components on the secondary side, i.e. the steam generators, by keeping the conductivity of the feed water as low as possible, and by avoiding making of any chemical additions to it.

A control program was developed in 1982-1983 to manage erosion-corrosion in the secondary system piping and components. Over the years, the amount of fittings to be inspected has been increased, based on the operation and inspection experience from Loviisa and other plants. Several modifications (repair, change of the components or piping and modifications of the system) have been made as a result of thickness measurements. At that time, 6-8 inspectors performed annually 200 inspection items per unit.

After the Surry-2 accident in 1986, the scope of the control program was enlarged , but it was still based on the operation experience and two-phase flow physics. After the enlargement, there were 300-400 inspected items per unit each year and 10-12 inspectors, who performed the annual inspections.

Due to shortcomings in the control program (the flow control orifices were not included in the program), the first guillotine pipe break of the feed water system occurred in 1990 at Unit 1. Erosion-corrosion adjacent to the flow control orifice in the feed water discharge line was incorrectly assumed to be lower than in the other fittings inspected during the previous outages (tees, 90 degree elbows, throttles etc.).

After the first guillotine pipe break, the scope of the control program was further increased to comprise about 600 items to be inspected every year in each unit. A computer program designed for the weak point analysis and for the inspection data management was implemented. Also, a special team of IVO experts was set up to evaluate the scope of the annual inspection program.

Based on the wall thickness measurements, the pipelines between the feed water pumps and the feed water discharge collectors were replaced with new pipes made of an erosion-corrosion resistant material. This renewal was completed at Loviisa 1 in 1992 and was scheduled to be performed at Loviisa 2 in 1994. Before that, in February 1993, however, in spite of the large efforts put on the improving the inspection program, a second guillotine break occurred at Loviisa 2. Self-assessment performed by IVO still revealed shortcomings in the inspection procedures and in the interpretation of the previous inspection results. Due to the very local nature of the wall thinning, it was found to be very important that the inspections be performed according to proper inspection procedures and instructions.

In order to tackle erosion-corrosion in the secondary system piping and components, several options have been implemented in recent years, such as, an exchange of a large extent of the piping and materials to a low alloy or stainless steel, an exchange of the components or improvements in the inspection techniques. One important feature to diminish erosion-corrosion was the change in the secondary water chemistry from neutral to alkaline water chemistry at Loviisa 2 in 1994 and at Loviisa 1 in 1995.

A more detailed overview on the feed water line breaks at Loviisa is given by Hietanen and Heltimoinen (1995).

In addition to erosion-corrosion, other forms of corrosion, such as pitting corrosion in the brackish sea water-cooled main turbine condenser tubes, have been a problem. The annual eddy current inspections performed since 1979 have revealed that the condition of the original copper-nickel tubes has been deteriorating rapidly, causing leaks and, consequently, tube plugging. In view of the high standards for condenser reliability, it was decided to retrofit the condensers. In both units, a condenser retrofit was carried out from 1985 to 1990. Both titanium and high alloyed stainless steel (254 SMO) were used as tube materials. Operational experience with both tube materials has so far been excellent.

Condition monitoring of electrical cables

Electrical cables inside the containment are qualified for 40 years of operation. However, this qualification includes many uncertainties. A condition monitoring program for cables is therefore necessary.

Only destructive methods are considered reliable. Samples of all the cable types used inside the containment are kept in a normal operational environment. Their insulation and sleeving materials are tested at intervals of 5 years to control any changes in their mechanical properties. Tensile strength and elongation before break are measured.

Now, after almost 20 years of operation, only minor deterioration has been found. The properties of most cables are still near the original ones. Considerable aging has been detected only in silicon rubber-insulated cables and cables located in an exceptionally hot operation environment. The aging tendency of the silicon cables was already detected in the qualification tests, and all the silicon rubber-insulated cables of safety-related circuits inside the containment were replaced. Our aim is to extend the life of cables by improving the environmental conditions.

Backfitting of valves

Loviisa Power Plant has some 140 valves which are classified as loss of coolant accident (LOCA) or main steam line break valves (later only LOCA-classified valves). IVO carried out a project from 1988 to 1996 to qualify the electric actuators as accident-classified. Some of the old valves did not fulfill the very strict sizing principles described below, and the valves, or parts of them, were replaced.

The main principle in the motor-operated valve is that in all LOCA-qualified valves (and some others, too) the actuator must not, under any circumstances, damage the valve, and the torque must always be sufficient to operate the valve. Following completion of the sizing check in the LOCA-program, IVO carried out a sizing check for all the other safety-related valves (about 400 positions).

Other life-limiting factors

Degradation of materials, caused by fatigue, corrosion, radiation and thermal effects, have been discussed above. In the following, some other life-limiting factors, or factors affecting possible life extension, will be discussed.

In general, components have been successfully replaced, such as batteries, computers, cables, valves and actuators. The plant computer of both units and the plant simulator computers have been replaced. New systems have been introduced to strengthen old plant designs: additional emergency feed water systems, essentially improved fire protection, operator support systems, etc., are all part of the task of modernizing existing plants.

Control rod drives were not replaced after five years, as recommended by the manufacturer. Instead, there is a program for periodic inspections and overhaul, which has indicated a need to change, periodically, only definite parts. Some new control rod drive systems have been acquired and installed into the system to permit overhaul of drives during power operation, but, so far, none of the original drives have been removed from service.

Diesel engines have not been considered for replacement, but a complete overhaul has been made one by one in an off-site factory, starting after 10 years of operation. During this year (1998), the last one will be overhauled. After this work, a future plan and a life estimation will be given to them.

A lot of pressure transmitters and containment wall instrument penetrations, among other things, have been replaced.

REFERENCES

Hietanen, O., Heltimoinen, M., A case study on plant safety self assessment: Feed water line breaks at Loviisa NPP. Proceedings of Topsafe 95, Budapest. Volume 1, p. 59 - 67.

Hytönen, Y., Ahlstrand, R., Kohopää, J., Šrachta, P., Annealing Loviisa 1 pressure vessel. Proceedings of PLIM+PLEX 1997, Prague. p. 337 - 344.

Laaksonen, J., Finnish view and perspective on VVER-440 ageing. Proceedings of PLIM+PLEX 1997, Prague. p. 1 - 7.

Rajamäki, P., Fatigue crack growth calculations for the pressurizer surge line nozzle and the steam generator feedwater nozzle. Specialists' Meeting on Sub-Critical Crack Growth. 14-18 May 1990, Moscow, USSR.

Saario, T., Törrönen, K., Ahlstrand, R., Ignatov, V. A., Fedorova, V., Rybin, V., Timofeev, B. T., Corrosion fatigue crack growth through cladding to base metal of a CrMoV reactor pressure vessel steel. SMIRT 1989, Los Angeles. Volume F, p. 135 - 139.

IMATRAN VOIMA OY
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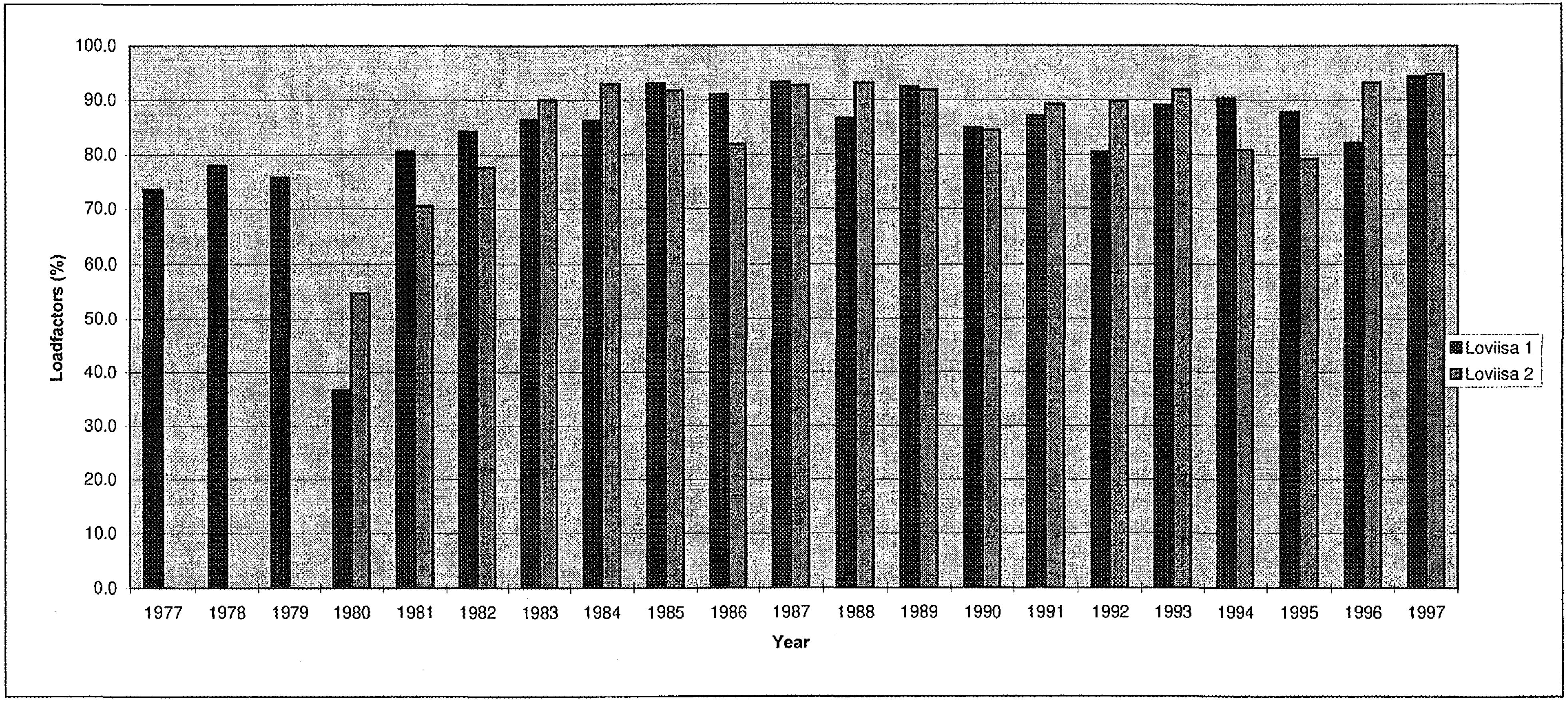
ATTACHMENT: LO1 & LO2 LOADFACTORS



	1977	1978	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997
Loviisa 1	73.5	78.0	75.8	36.7	80.6	84.2	86.4	86.2	93.0	91.0	93.2	86.7	92.4	84.9	87.1	80.5	89.0	90.3	87.8	82.2	94.3
Loviisa 2				54.7	70.5	77.7	90.0	92.9	91.7	81.9	92.6	93.1	91.8	84.5	89.2	89.8	91.8	80.9	79.2	93.2	94.7