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WP-Cave – assessment of feasibility, safety and development potential

Swedish Nuclear Fuel and Waste Management Company, Stockholm

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WP-CAVE - ASSESSMENT OF FEASIBILITY, SAFETY AND DEVELOPMENT POTENTIAL

Swedish Nuclear Fuel and Waste Management Company, Stockholm

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WP-CAVE –

ASSESSMENT OF FEASIBILITY, SAFETY AND DEVELOPMENT POTENTIAL

ABSTRACT

According to SKB R&D-programme 1986, alternative disposal methods will be investigated to provide a basis for selecting a site and a repository system for the Swedish spent nuclear fuel.

The present report is a comparison between the WP-Cave and the reference concept KBS-3.

The evaluation was made jointly by the SFG (the integrated performance group) and the WP-Cave project.

The task of the group has been to recommend to SKB which of the two alternative repository concepts should be prioritized in the future R&D efforts.

The comparison between the WP-Cave and KBS-3 concept has resulted in the following conclusions:

- Both concepts are judged to be able to provide adequate safety.
- A utilization of the potential of the WP-Cave requires, however, extensive development in areas where the current state of knowledge and available data are incomplete.
- The higher temperatures in the WP-Cave lead to greater uncertainty as to long-term performance. Reducing this uncertainty would require many years of research and substantial resources.
- Both repositories, including the barriers they incorporate, could be built with a normal adaptation of available technology.
- It is not possible to say today whether it would be simpler to find suitable sites for one design or the other.
- The WP-Cave is considerably more expensive.

A future research direction based on a concentrated emplacement of spent fuel along the lines of the WP-Cave is therefore judged to entail greater uncertainty as regards the possibilities of achieving acceptable safety and to require greater resources for research and development, at the same time as the costs of building the repository would be higher.

The studies of the WP-Cave as an integral system should therefore be discontinued. The research should be focused on distributed systems with lower temperatures, in accordance with the basic KBS-3 concept.

Certain barrier designs in the WP-Cave could also be utilized in repository designs with lower temperatures, for example the reduction potential of the steel canisters and the hydraulic cage's diversion of groundwater. Studies within these areas are being conducted within SKB and should continue.

PREFACE

According to SKB R&D-programme 1986, alternative disposal methods will be investigated to provide a basis for selecting a site and a repository system for the Swedish spent nuclear fuel.

The present report is a comparison between the WP-Cave and the reference concept KBS-3.

The evaluation was made jointly by the SFG (the integrated performance group) and the WP-Cave project

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Comparison between Development Potential for the WP-Cave and KBS-3

One of the objectives of the SKB is to build a final repository for Sweden's spent nuclear fuel. The feasibility of a safe final disposal scheme was analyzed in 1983 in the KBS-3 Report. In 1984, the Government found that the reported method was acceptable from the viewpoint of safety. Since then, the purpose of SKB's work has been to gather a comprehensive body of data on which to base the selection of a site for the repository and a suitable design of the facility and its engineered barriers.

According to present-day plans, the site and design of the repository will be chosen at the turn of the century, and a siting application will be submitted in 2003.

In order to gather a comprehensive body of data on which to base these decisions, alternative repository designs must be examined and compared. Furthermore, choices between very different alternatives must be made at an early stage so that research and development can be steered onto the right track.

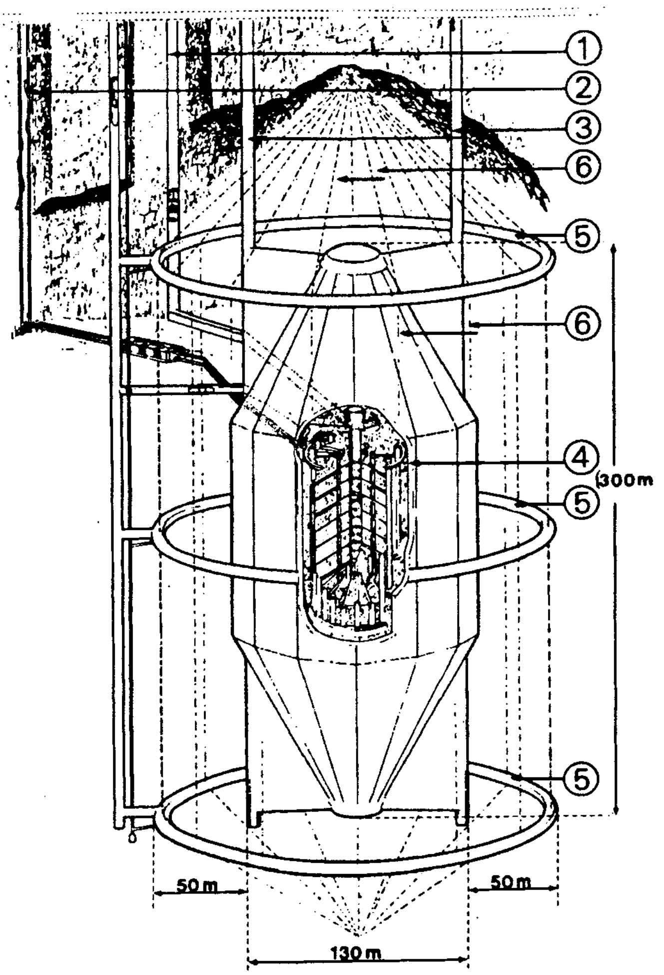
WP-Cave is the name of a concept for a final repository for spent nuclear fuel originally developed by WP-System as an alternative to the KBS concept. A brief description of the design of the WP-Cave on which the performance assessments in this study have been based is given below. Unless otherwise clearly stated, this is the design that is referred to in this study by WP-Cave. Naturally, the general concept of the WP-Cave can take a variety of different forms in terms of size and detailed design.

The fundamental characteristic difference between the two concepts is that in the WP-Cave, the encapsulated waste is emplaced in a relatively small, concentrated rock volume that is surrounded by thick clay barrier and a hydraulic cage. KBS-3 is based on the principle of distributing the fuel canisters over a larger rock volume and surrounding each canister with its own clay barrier. The purpose in the WP-Cave is to take advantage of a thick clay barrier and thereby be relatively less dependent on the characteristics of the surrounding rock. However, concentrating the waste in a smaller volume leads to higher temperatures near the waste canisters. The idea in KBS-3 was to keep temperatures down by distributing the waste so as to minimize any disturbance of the natural conditions in the rock and to choose engineered barriers with proven, very good long-term performance in the natural rock environment.

Since the difference in temperature is important for the research that has to be conducted, it was deemed important in preparation for the 1989 research programme to perform an evaluation of which of the ideas should be given priority from now on.

During the period 1987—1989, the WP-Cave design has been analyzed and compared with a reference design based on the KBS-3 concept. Aspects of importance in the comparison were found to be:

- safety and potential for development,
- confidence in the performance assessments,
- technical feasibility and
- construction costs.



1. Transportation shaft
2. Ventilation shaft
3. Main shaft for excavation and refilling of slot
4. Bentonite-sand barrier with thickness of 5 m
5. Drift for hydraulic cage
6. Drillhole for hydraulic cage

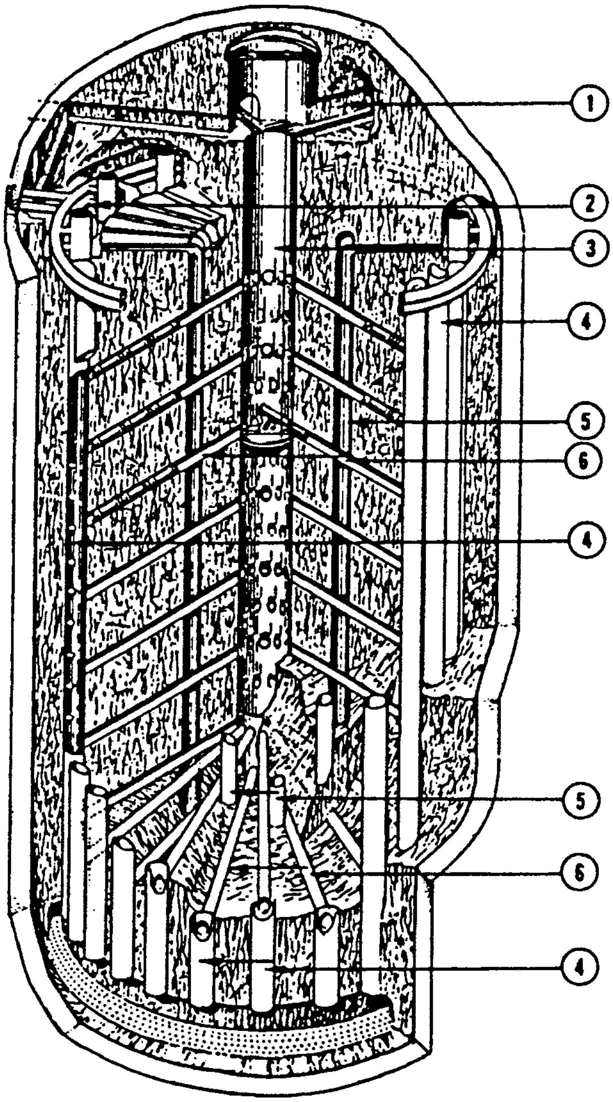
Figure 1. Overview of analyzed WP-Cave design.

Design of the WP-Cave

The design used as a basis for the present analysis is shown in Figure 1. The spent fuel is placed in the center of the cave, surrounded by a rock barrier. Around this, a slit is excavated and backfilled with a mixture of bentonite clay and sand. Outside the bentonite-sand barrier a set of horizontal drifts are placed, connected by vertically bored drainage channels, to constitute a hydraulic equivalent of a Faraday cage.

The size is adjustable in principle. The dimensions of a cave designed to host spent fuel corresponding to 1 100 tonnes of uranium are shown in Figure 1. This particular variant is referred to as WPC 1100.

The inner parts of the cave, where the spent fuel is emplaced, is shown schematically in Figure 2. From the central shaft storage channels extend with a 30 degree downward inclination. There is room for twelve such channels on each level. Their diameter is made sufficiently large to allow cooling air to pass the individual canisters.



1. Bulk storage room
2. Heat exchange
3. Central storge shaft
4. Outer ventilation shaft
5. Inner ventilation shaft
6. Canister channel

Figure 2. Central part of repository.

During the operational phase, cooling air circulates downward in the outer, vertical shafts and upward in the inner shafts. The air is recirculated via a heat exchanger.

In the WPC 1100, two canisters are placed in each channel. Each canister contains sixteen BWR or five PWR assemblies, corresponding to some 3 tonnes of uranium per canister. This gives a total of 384 canisters and sixteen levels. The height of the inner part is approximately 110 m, given 3.5 m between each level.

The canister is assumed to be made of steel with a low carbon content. All storage channels and shafts are lined with steel to prevent dust from accumulating in the ventilation system. Concrete is not used since it does not provide any benefits as to safety in the long term.

The rock mass between the storage section and the bentonite-sand barrier prevents the clay from being exposed to high temperatures. The requirement is that the temperature should not exceed 80 degrees. The overall dimensions are adjusted accordingly (Figure 1), on the conservative side. The number of canisters in each channel (two) is determined by the requirement that the temperature on the surface of the canisters should not exceed 150 degrees. If three canisters were to be placed in each channel, the surface temperature would exceed 200 degrees, whereas the temperature in the clay would still not exceed 80 degrees.

The cave is closed, after a hundred year operational period, by backfilling all open spaces inside the clay barrier with finely ground sand (< .1 mm) and water.

Safety and Potential for Development

The design of the internal parts of the WP-Cave with a large inner rock mass of high permeability is such that the transport of substances dissolved in water from the repository to the host rock is dominated by advection (flowing water), despite the fact that the repository is surrounded by a thick clay barrier.

As a result of the choice of steel for the canister material, along with other design parameters, the ability of the repository to prevent the release of radionuclides to the environment is dependent entirely on the clay barrier. Consequently, the overall safety of the repository is also highly dependent on the quality of this single barrier.

In the KBS-3 design, material transport takes place by diffusion. This, together with the use of copper canisters and lower temperatures, has enabled the repository to be designed in such a way that the prevention of radionuclide release is distributed more evenly between canister life, low solubility and transport resistance. These advantages would be gained also in the WP-Cave alternative if long-lived canisters would replace the short-lived steel canisters.

Calculations show that it can be difficult, under certain pessimistic but conceivable conditions, to guarantee that radiological requirements can be met with the present-day design of the WP-Cave. However, there is nothing to prevent the WP-Cave from being modified so that an acceptable level of safety can be achieved. But this requires a further improvement of the present barrier system and further model development and data collection to permit better assessment of barrier performance at the temperatures in question. If copper is substituted for steel as canister material a significant improvement is made in the WP-Cave barrier system.

It has been difficult to evaluate the performance of the hydraulic cage of drilled water pathways around the cave. The purpose of the cage is to reduce the hydraulic gradient over the repository. A favorable consequence is that the natural flow of groundwater can be led around the repository thanks to the short-circuiting of the fracture systems, so that it does not reach the bentonite barrier. An unfavorable consequence is that this short-circuiting also means that all the radioactivity that leaks out of the repository will also be conducted up to the biosphere via the fastest transport channel that reaches the cage. A further difficulty is assessing the long-term stability of the newly-created interconnection pathways under the existing temperature gradient. Further model development is required here as well.

The comparison between the designs with regard to safety and potential for further development gives no reason to dismiss the WP-Cave. However, in order to exploit the potential of the WP-Cave concept, a great deal of effort must be devoted to model development in areas where our present-day understanding and existing data are incomplete.

Confidence in the Performance Assessments

A fundamental requirement for obtaining a licence to construct the final repository is that it can be demonstrated that the uncertainties associated with the safety analysis are not so great that they influence the assessment of whether the repository meets the safety requirements.

Every assessment of the long-term performance of a final repository, no matter its design, contains considerable uncertainties. However, the assessment of the performance of final repositories based on the two alternatives being considered shows that the higher initial temperature in WP-Cave, in addition to the effects mentioned above, also results in increased uncertainty regarding the solubility of the radionuclides, regarding the performance of the bentonite barrier as it is affected by silica precipitation and regarding the long-term performance of the hydraulic cage. These uncertainties stem both from uncertainty regarding which chemical processes are dominant at higher temperatures, and from a lack of data.

The comparison with respect to confidence in the performance assessments shows that the higher temperatures in WP-Cave give rise to a higher uncertainty regarding long-term performance. An internationally coordinated programme lasting many years and requiring substantial resources would be required to reduce this uncertainty.

All waste emplaced in the Swedish bedrock is in principle accessible as long as the canisters are intact and the location of the repository is known. The cost of retrieving the waste will depend on how it is encapsulated and the extent to which the repository has been backfilled and sealed.

Confidence in the safety of the repository has sometimes been coupled to the possibility of monitoring the repository over a long period of time. The safety value of a long period of inspection and supervision is, however, doubtful, since this can often only be achieved by postponing the completion of the repository's passive safety systems.

Technical Feasibility

Nothing has emerged from the studies to cast doubt on the assumption that both the KBS-3 and the WP-Cave repositories, including the barriers they incorporate, can be built with a normal adaptation of available technology.

There are, however, differences between the two systems. WP-Cave requires a predetermined geometric layout of the facility. The KBS-3 repository, on the other hand, has been designed for the expressed purpose of being flexibly adaptable to local rock conditions. Both tunnel routes and the location of the deposition holes can be adapted to conditions measured during the construction of the repository, as long as a minimum spacing of the canisters is observed. Thus, the KBS-3 design makes it possible to exploit the characteristics of the rock as a natural barrier against the dispersal of radioactive materials to a higher degree.

The geological investigations conducted in various parts of the country have to some extent changed the picture of how frequently horizontal or near-horizontal major fractures zones occur in the Swedish bedrock.

Such zones were previously believed to be uncommon. Today it has been found that the average distance between these zones is 200—300 metres in some places. The consequence is that it may be more difficult than previously believed to find sites for repository designs which, like WP-Cave, require a vertical extent of about 400 m. From this viewpoint as well, the flexibility of a design where the canisters can be spread out is an advantage.

The judgement of whether a given repository site is suitable must accordingly take into account both requirements on rock volumes of suitable quality and requirements on the geometry of the repository. In this light, it is doubtful whether the original goal of the WP-Cave design, namely to facilitate siting of the repository, can be achieved.

Considering the overall state of knowledge today, there are no grounds for asserting that it would be easier to find suitable sites for one design or the other.

Costs

The cost of a complete system for the handling and final disposal of spent nuclear fuel based on the WPC 1100 design has been calculated to be SEK 44 billion, compared with SEK 28 billion for a system based on KBS-3. The WP-Cave design is thus considerably more expensive.

An assessment has further been made of the potential for cost reductions in WP-Cave. The only way to obtain significant cost reductions would be to utilize larger, and thereby fewer, units or to reduce the thickness or the quality of the bentonite layer. However, both of these alternatives would have repercussions on the safety assessments, via the temperature and via the capacity of the bentonite barrier to reduce the leakage of radionuclides.

The quality and reliability of the engineered barriers, regardless of which design alternative is chosen, is often closely related to the cost of building them. Significant safety advantages should therefore be required in order to compensate for the choice of a considerably more expensive system as the main direction of future research within SKB.

Conclusions

The comparison between the WP-Cave and KBS-3 concept has resulted in the following conclusions:

- Both concepts are judged to be able to provide adequate safety.
- A utilization of the potential of the WP-Cave requires, however, extensive development in areas where the current state of knowledge and available data are incomplete.
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SAMMANFATTNING

Jämförelse mellan utvecklingspotentialen för WP-Cave och KBS-3

SKB har bl a till uppgift att bygga ett slutförvar för Sveriges använda kärnbränsle. Genomförbarheten av en säker slutförvaring redovisades 1983 med KBS-3-rapporten. Regeringen fann 1984 att den redovisade metoden var acceptabel ur säkerhetssynpunkt. Sedan dess syftar SKBs arbete till att bygga upp ett allsidigt beslutsunderlag för att välja plats för förvaret samt lämplig utformning för anläggningen och dess tekniska barriärer.

Enligt nuvarande planer kommer plats och utformning att väljas vid sekelskiftet, och en lokaliseringsansökan att inlämnas år 2003.

För att man ska kunna få ett allsidigt underlag måste alternativa förvarsutformningar granskas och jämföras. Vidare måste val mellan alternativ som är mycket olika göras tidigt för att forskning och utveckling skall kunna styras på ett målinriktat sätt.

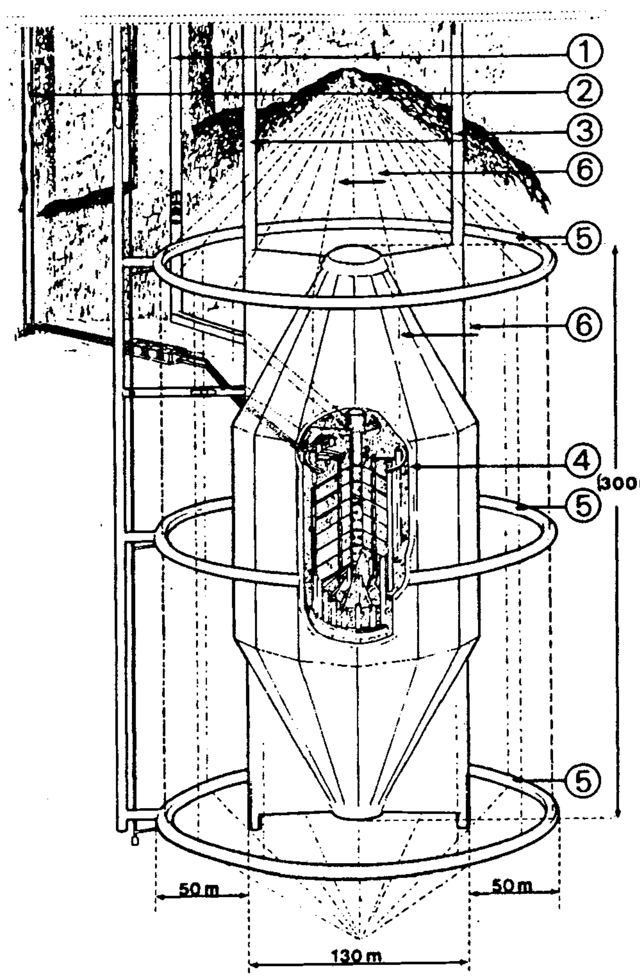
WP-Cave är en utformning av ett slutförvar för använt kärnbränsle ursprungligen utarbetad av WP-System som ett alternativ till KBS-3. En kortfattad beskrivning av den utformning av WP-Cave som legat till grund för funktionsanalyserna i denna studie framgår av nästa underavsnitt. Om ej annat anges så åsyftas i fortsättningen med WP-Cave denna speciella utformning. Den generella idén WP-Cave kan givetvis ges en mängd olika utformningar vad avser storlek och detaljutformning.

Den grundläggande karakteristiska skillnaden mellan de två koncepten är att i WP-Cave placeras det inkapslade avfallet i en relativt liten koncentrerad bergvolym som omges av en tjock lerbarriär och en hydraulisk bur. KBS-3 bygger på principen att fördela bränslekapslarna över en större bergvolym och omge varje kapsel med sin egen lerbarriär. Syftet i WP-Cave är att kunna tillgodoräkna sig en tjock lerbarriär och därigenom bli relativt sett mindre beroende av det omgivande bergets egenskaper. Koncentrationen till mindre volym leder dock till högre temperaturer närmast avfallskapslarna. Syftet i KBS-3 är att genom relativt gles deponering hålla låga temperaturer och få minsta möjliga förändring av de naturliga förhållandena samt välja tekniska barriärer med dokumenterad mycket god långtidsfunktion i den naturliga bergmiljön.

Då bl a skillnaden i temperatur är viktig för den forskning som behöver genomföras bedömdes det viktigt att inför 1989 års forskningsprogram göra en utvärdering av vilken av ideerna som i fortsättningen skulle prioriteras.

Under perioden 1987 — 1989 har WP-Cave-utformningen analyserats och jämförts med en referensutformning enligt KBS-3. Frågor av vikt för jämförelsen visade sig vara

- säkerhet och potential för vidareutveckling,
- tilltro till funktionsanalyserna,
- teknisk genomförbarhet samt
- anläggningskostnader.



1. Transportschakt
2. Ventilationsschakt
3. Huvudschakt för utsprängning och återfyllning
4. Bentonit-sand barriär med 5 m tjocklek
5. Ort för hydrauliska buren
6. Borrhål för hydrauliska buren

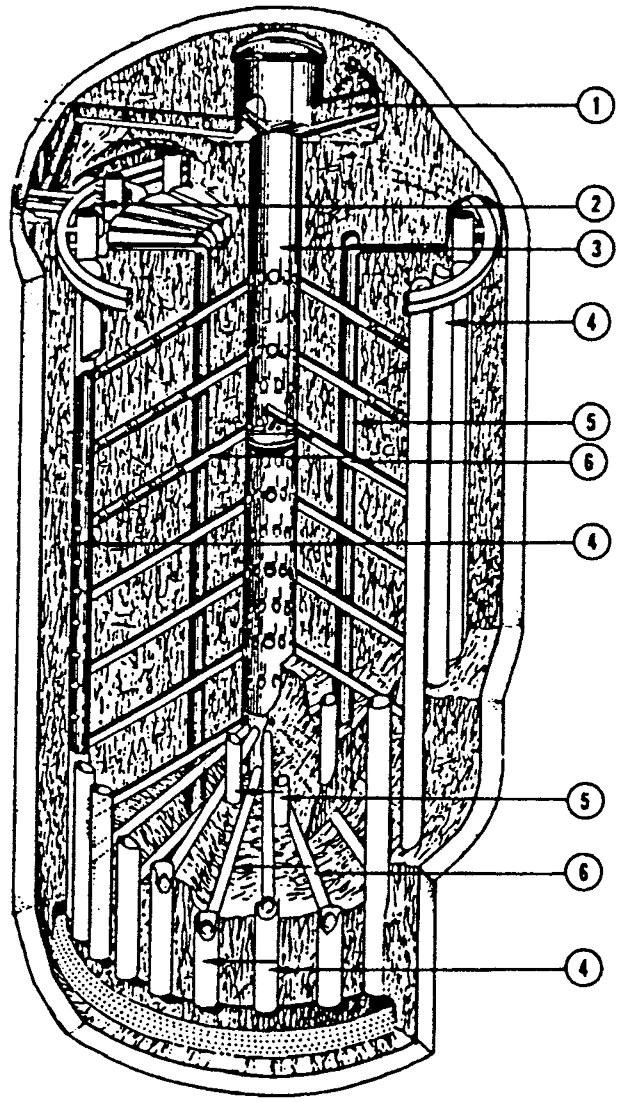
Figure 1. Översikt av den analyserade utformningen av WP-Cave.

Utformningen av WP-Cave

Den studerade utformningen av WP-Cave framgår av Figur 1. Det använda bränslet placeras i centrum av anläggningen och omges av en barriär av berg. Runt denna sprängs en slits ut, som återfylls med en blandning av bentonit och sand. Utanför bentonit-sand-barriären arrangeras en sk hydraulisk bur, vilken består av horisontella orter sammanbundna av vertikalt borrade dräneringskanaler.

Storleken kan ändras. En WP-Cave som beräknas kunna rymma bränsle motsvarande 1 100 ton uran, en sjundedel av det svenska programmet, har de mått som framgår av Figur 1. Denna storlek benämns WPC 1100.

Den centrala delen av förvaret, där bränslet lagras, visas skissartat i Figur 2. Från ett schakt i mitten sträcker sig lagringskanaler radiellt utåt men i ca 30° lutning neråt. På varje nivå finns det plats för tolv sådana kanaler. Deras diameter är så pass mycket större än bränslekapslarnas, att plats finns för kylluften att passera.



1. Utrymme för stora avfallskollin
2. Värmeväxlare
3. Centralschakt
4. Yttre ventilationsschakt
5. Inre ventilationsschakt
6. Kapselkanal

Figur 2. WP-Cave förvarets lagerdel.

Kylluften cirkulerar neråt i de yttre vertikala schakten, igenom kapselkanalerna och uppåt i de inre schakten. Efter värmeväxling återcirkuleras luften.

I WPC 1100 lagras två kapslar i rad i varje kanal. Varje kapsel rymmer 16 BWR-element eller 5 PWR-element. Räknat på BWR-element är vikten per kapsel ca 2,9 ton U. Detta betyder att 1 100 ton U fordrar 16 nivåer (384 kapslar). Höjden i det inre förvaret blir härvid ca 110 m, om avståndet mellan varje nivå av lagringskanaler är 3,5 m i vertikalled.

Kapseln förutsättes vara av stål med låg kolhalt.

Alla lagringskanaler och schakt förutsättes bli klädda med stålinfodring, för att trygga en dammfri miljö i alla ventilationskanaler. Betong används ej, eftersom detta material ej bedöms medföra någon speciell fördel ur säkerhetssynpunkt på lång sikt.

Bergmassan mellan det centrala lagerutrymmet och bentonit-sandbarriären skall skydda bentoniten från att bli alltför varm. Kravet är att temperaturen ej stiger till mer än 80°C i bentoniten. Detta har resulterat i de presenterade måtten i Figur 1, vilka dock är något konservativt tilltagna. Förutsättningen vid bestämningen av antalet kapslar i varje lagringskanal är att temperaturen ej stiger till mer än 150°C på kapslarnas yta. Skulle tre kapslar placeras i varje lagringskanal stiger temperaturen på kapslarnas ytor till mer än 200°C, men fortfarande ej över 80°C i bentoniten.

Vid förslutning, efter ca 100 års ventilation och övervakning, återfylls allt öppet utrymme innanför bentonit-sandbarriären med finmald sand, finare än 0,1 mm, och vatten.

Säkerhet och potential för vidareutveckling

Utformningen av de inre delarna av WP-Cave med en stor inre bergkropp, som har hög vattengenomsläpplighet, medför att transporten av i vatten lösta ämnen från förvaret till omgivande berg domineras av advektion (strömmande vatten). Detta trots att förvaret omsluts av en tjock lerbarriär.

Valet av stål som kapselmaterial och dimensioneringen i övrigt har samtidigt som följd att begränsningen av radionuklidutsläppet till omgivningen helt beror på lerbarriären. Som följd härav blir också den totala säkerheten starkt beroende av denna enda barriärs kvalitet.

I KBS-3 utformningen sker materialtransporten med diffusion. Detta tillsammans med kapslar av koppar och lägre temperaturer har medgett att förvaret kunnat dimensioneras så att begränsningen i utsläpp fördelas mera jämbördigt mellan kapsellivslängd, svårlöslighet och transportmotstånd.

Genomförda beräkningar visar på att det kan vara svårt att, under vissa ogynnsamma men dock tänkbara förhållanden, kunna garantera att strålskyddskraven kan uppnås med den nuvarande utformningen av WP-Cave. Det finns emellertid ingenting som motsäger att WP-Cave skulle kunna modifieras så att en acceptabel säkerhet skulle kunna erhållas. Detta kräver emellertid både en vidare utveckling av barriärsystemet, där bl a stålet ersätts med koppar, och en vidare utveckling av modeller och dataunderlag för att bättre kunna utvärdera barriärfunktionerna vid aktuella temperaturer.

Funktionen av den sk hydrauliska buren av uppborrade vattenvägar runt WP-Cave har varit svår att utvärdera. Syftet med buren är att reducera den hydrauliska gradienten över förvaret. En gynnsam följd är att den naturliga grundvattenströmmen genom kortslutningen av spricksystemen kan ledas runt förvaret istället för att nå fram till bentonitbarriären. Ogynnsamt är att kortslutningen samtidigt innebär att all den aktivitet som läcker ut ur förvaret också kommer att föras upp till biosfären via den snabbaste transportkanal som når buren. En ytterligare svårighet är att bedöma långtidsstabiliteten i de nyskapade kontaktvägarna under aktuella temperaturgradienten. Även här krävs en fortsatt modellutveckling.

Jämförelsen mellan utformningarna vad gäller deras säkerhet och potential för vidareutveckling ger inte någon anledning att avfärda WP-Cave. Ett utnyttjande av WP-Caves potential kräver dock en omfattande insats för modellutveckling på områden där förståelsen och dataunderlaget idag är ofullständigt.

Tilltron till funktionsanalyserna

Ett grundläggande krav för tillståndet att bygga ett slutförvar är att det kan visas att de osäkerheter som vidlåter säkerhetsanalysen inte är så stora att de påverkar bedömningen av om förvaret uppfyller säkerhetskraven.

Varje bedömning av långtidsfunktionen för ett slutförvar, vilken utformning den än har, innehåller avsevärda osäkerheter. Vid analysen av hur slutförvar enligt de båda alternativen fungerar visar det sig dock att den högre temperaturen i WP-Cave, förutom tidigare nämnda effekter, också ger en ökad osäkerhet i radionuklidernas löslighet, i bentonitbarriärens funktion med avseende på betydelsen av silikatutfällning och i långtidsfunktionen för den hydrauliska buren. Dessa osäkerheter föranleds både av osäkerhet om vilka kemiska processer som dominerar vid högre temperatur och av ett bristande dataunderlag.

Jämförelsen vad gäller tilltron till funktionsanalyserna visar att de högre temperaturerna i WP-Cave medför en ökad osäkerhet i långtidsfunktionen. För att reducera denna osäkerhet erfordras en mångårig, resurskrävande och internationellt samordnad insats.

Allt avfall som placeras i svensk berggrund är i princip åtkomlig så länge kapslarna är intakta och förvarets läge är känt. Kostnaden för att återta avfallet kommer att vara beroende av hur inkapslingen är gjord och till vilken grad förvaret återfyllts och förseglats.

Tilltron till förvarets säkerhet har ibland kopplats till möjligheten att under lång tid påverka förvaret. Säkerhetsvärdet av en lång period av inspektion och kontroll är emellertid tveksamt, då möjligheten ofta endast kan erhållas genom att färdigställandet av förvarets passiva säkerhetssystem skjuts upp.

Teknisk genomförbarhet

Ingenting har framkommit vid studierna som ifrågasätter bedömningen att båda förvaren, inklusive de ingående barriärerna, kan byggas med en normal anpassning av tillgänglig teknik.

Det finns dock skillnader mellan de båda systemen. WP-Cave kräver en förutbestämd geometrisk utformning av anläggningen. KBS-3-förvaret har däremot utformats just med syftet att flexibelt kunna anpassas till lokala bergförhållanden. Både tunnelsträckningar och placeringen av deponeringshålen kan anpassas till förhållanden som uppmäts vid utbyggnaden av förvaret bara ett minimiavstånd mellan kapslarna iakttas. KBS-3-utformningen gör det således möjligt att i högre grad utnyttja bergets egenskaper som naturlig barriär mot spridning av radioaktiva ämnen.

De geologiska undersökningarna i olika delar av landet har i viss mån förändrat bilden av hur ofta horisontella eller nära horisontella större sprickzoner förekommer i den svenska berggrunden.

Förut ansågs att dessa var ovanliga, idag visar det sig att medelavståndet mellan dem på många ställen är 200 à 300 m. Konsekvenserna av detta är att det kan vara svårare än vad man tidigare trott att hitta lokaliseringar för förvarsutformningar som liksom WP-Cave kräver en vertikal utsträckning av ca 400 m. Även ur denna synpunkt är flexibiliteten i en utformning där kapslarna kan spridas ut en fördel.

Bedömningen av om en viss förläggningsplats duger måste således ta hänsyn både till krav på bergvolymer av lämplig kvalitet och krav på förvarets geometri. Det är således tveksamt huruvida den ursprungliga målsättningen med WP-Cave-utformningen, nämligen att underlätta slutförvarets lokalisering, har uppnåtts.

Med hänsyn till kunskapsläget i stort är det idag inte möjligt att hävda att det skulle vara enklare att hitta lämpliga förläggningsplatser för den ena utformningen eller den andra.

Kostnader

Kostnaden för ett fullständigt system för slutförvaring av använt kärnbränsle baserat på WPC 1100 har beräknats till 44 miljarder kr jämfört med 28 miljarder för ett system baserat på KBS-3. WP-Cave-utformningen är således avsevärt dyrare.

En bedömning har vidare gjorts av potentialen för kostnadsreduktioner i WP-Cave. De enda sätt på vilka signifikant minskade kostnader skulle kunna erhållas vore att utnyttja större, och därmed färre, enheter eller att reducera tjockleken av bentonitskiktet eller dess kvalitet. Båda vägarna kopplar emellertid till säkerhetsbedömningarna, via temperatur och via bentonitbarriärens förmåga att reducera utläckaget av radionuklider.

De tekniska barriärernas kvalitet och säkra funktion, oavsett utformningsalternativ, är ofta nära kopplad till kostnaden för att bygga upp dem. Uppenbara säkerhetsfördelar bör därför krävas för att kompensera att ett väsentligt dyrare system väljes som huvudinriktning för den framtida forskningen inom SKB.

Slutsatser

Jämförelsen mellan WP-Cave- och KBS-3-alternativen har resulterat i följande bedömning:

- Båda koncepten bedöms kunna ge en tillräcklig säkerhet.
- Ett utnyttjande av WP-Caves potential kräver dock en omfattande utveckling på områden där förståelsen och dataunderlaget idag är ofullständigt.
- De högre temperaturerna i WP-Cave medför en ökad osäkerhet i långtidsfunktionen. För att reducera osäkerheten erfordras en mångårig resurskrävande insats.
- Båda förvaren, inklusive de ingående barriärerna, bedöms kunna byggas med en normal anpassning av tillgänglig teknik.
- Det är inte idag möjligt att säga om det skulle vara enklare att hitta lämpliga förläggningsplatser för den ena utformningen eller den andra.
- WP-Cave är avsevärt dyrare.

En framtida forskningsinriktning på koncentrerad inlagring av använt bränsle enligt WP-Cave bedöms därför innebära större osäkerhet rörande möjligheterna att åstadkomma acceptabel säkerhet och kräva större resurser för forskning och utveckling samtidigt som kostnaderna att bygga förvaret skulle bli högre.

Studierna av WP-Cave som ett sammanhållet system bör därför avslutas. Forskningen bör inriktas på distribuerade system med lägre temperatur enligt grundtankarna bakom KBS-3.

Vissa barriärutformningar i WP-Cave skulle även kunna utnyttjas i förvarsutformningar med lägre temperatur, t ex stålkapslarnas reduktionspotential och den hydrauliska burens förbiledning av grundvattnet. Studier inom dessa områden pågår inom SKB och bör kunna fortsätta.

1 GENERAL BACKGROUND

1.1 STUDIES OF ALTERNATIVE DISPOSAL METHODS

In 1984 Swedish Government accepted the conclusions in the KBS-3 report that it is possible to manage and dispose of the spent fuel from the nuclear power production in Sweden. The R&D-programme 1986 was published in 1986 /1-1/. The report presented the long-term planning of how a final disposal facility should be arranged, sited and built in Sweden. This was accepted by the Government in 1987.

The present studies of alternative disposal methods are intended to broaden the decision base for the selection of a site and a system for the planned repository in Sweden. By evaluating the technical feasibility of various alternative concepts, their safety potential and their associated costs, a safe and cost-effective disposal concept can be selected in Sweden.

The goals of the studies of the alternatives are:

- to find out whether obviously beneficial aspects have been missed out in earlier studies,
- to evaluate the technical feasibility of the alternatives, the requirements they put on the rest of the back-end system and their safety potential,
- to enable the research and development effort to be focused in a direction that at the same time will
  - give high potential for safe repositories in Sweden,
  - preserve a high degree of freedom to change the direction especially during the research and development phase, when the basis for ranking the alternatives is weak, and
  - be effective with regard to cost and utilization of scientific personnel.
- to further improve the knowledge base on which the future adaption of the repository to the natural characteristics of the site and to the acceptance criteria of the society will be made.

According to "R&D-programme 1986" the focusing of the investigations will take place according to the following time table:

1989	Selection of a site for the hard rock laboratory. In order for this laboratory to provide relevant data, this also entails a decision in principle regarding a preferred rock-type and repository concept.
1991	Selection of sites for the detailed geological investigations. The selection process requires that some basic characteristics of the repository on the site have been defined, for instance the approximate depth, the area requirements, required homogeneity of the rock, etc.
95/96	Selection of materials and configuration of the technical barriers. The selection will be based on geological parameters existing at those sites selected for the detailed geological investigations.
98/99	The final selection of one site and an appropriate repository design.

1.2 ROLE OF THE SFG

The goals as defined above require a systematic effort to evaluate performance, to compare the concepts and to assign priorities to the available alternatives. For this purpose the SFG was formed in the beginning of 1987 (SFG = Samfunktionsgruppen, the integrated performance group). It's role is to evaluate and compare the various alternatives for final disposal and/or safety barriers as a basis for ranking the alternative approaches and focusing the R&D-activities.

Evaluation by the SFG of the various alternatives will be carried out in cooperation with the project leader responsible for the investigations of the alternative.

Since the KBS-3 concept was analyzed and reviewed in great detail it will be used as a reference alternative. An effort will be made to compare the alternatives on the basis of expected performance and not pessimistic cases.

The KBS investigations indicated that there are many places in Sweden where a repository of the KBS-3 type could be sited to provide an acceptable level of safety. The investigations also showed that there are a great number of alternative layouts of the repository and of configurations for the barrier system in KBS-3 whose technical feasibility seems to be equally good.

1.3 THE WP-CAVE/KBS-3 COMPARISON

The present comparison between the WP-Cave concept and the reference concept KBS-3 is basically a comparison between a compact repository allowing high temperatures and substantial change of the near field environment with a distributed repository trying to preserve the natural conditions in the host rock.

Another consequence of the compact design of the WP-Cave is that the repository must be built with a given geometry regardless of variations in local rock quality as soon as the site has been selected. Consequently, the safety barriers in the WP-Cave mainly consist of engineered barriers.

The concept of distributing the waste in canisters over a larger area makes it possible to adjust the geometry of the repository to local rock quality and thereby to utilize the barrier potentials of the geosphere to a greater extent.

Since the factors mentioned above also have a bearing on the necessary R&D, on how the rock mass of a repository should be characterized and how to design and make relevant tests and experiments in the Hard Rock Laboratory, it was found advantageous to make an early choice between these basic concepts.

To provide the information necessary for this comparison the WP-Cave study within SKB was started in 1986.

1.4 OUTLINE OF THE REPORT

Chapter 2 contains a description of the factual information that forms the basis of the evaluations and comparisons. **Section 2.1** accounts for the early history of the Cave. **Section 2.2** covers the reference design and layout of the repository, possibilities for alternative designs and materials are discussed. **Section 2.3** discusses design factors important to performance and safety. **Section 2.4** covers the cost aspects by presenting the reference system for the cost calculations, discussing the alternatives and finally summarizing the factors of importance for the cost calculations.

In **Section 2.5** the results and the conclusions of the performance assessments are presented. The first subsection presents the assessment strategy and the second presents the low-flow-through case selected for this assessment and the assessment results. The consequences of possible variations in the system configuration are discussed and the final subsection gives a summary of features and phenomena that are important to the safe performance of the repository. **Section 2.6** summarizes conclusions regarding the technical and economical feasibility of the WP-Cave and the level of safety that can be achieved by it.

Chapter 3 compares the KBS-3 concept and the WP-Cave concept with regard to safety and the level of confidence that can be expected to be achieved in the assessment. After a short introduction in 3.1, **Section 3.2** discusses the 100-year-long active cooling phase. **Section 3.3** covers the consequences of the geometrical layout of the WP-Cave and **Section 3.4** comments on the differences between KBS-3 and WP-Cave with regard to physical structure. **Section 3.5** deals with the temperatures that result from the compact design of the WP-Cave.

Section 3.6 presents some major differences between the barriers in WP-Cave and KBS-3 and discusses their importance for safety and their consequences with regard to provability and uncertainty. In **Section 3.7** the risk of gas generation in the repository and its impact on safety are discussed.

Section 3.8 provides a short overview of the consequences of some external events and in **Section 3.9** there is a discussion of uncertainties and confidence issues, both from a general point of view and for the specific comparison at hand.

Chapter 4 presents the conclusions and the recommendation of the SFG on the future direction of the R&D with regard to the comparison made.

2 TECHNICAL AND ECONOMICAL FEASIBILITY OF THE WP-CAVE CONCEPT

2.1 EARLY HISTORY

The WP-Cave design was originally conceived in 1976 as a spin-off of development work concerning larger underground construction including subsurface siting of nuclear power plants and underground storage of large quantities of hydrocarbon products. The name of the concept is partly taken from the name of the innovator, the consulting and construction company WP-System AB. The first design of the WP-Cave looks very much different from the one of today, but is based on the same main features.

When the KBS project started in December 1976, conceptual ideas and designs where invited from various groups in Sweden. One of the submitted ideas was an early variant of the WP-Cave, a large cavern supported by a reinforced concrete rib-structure in the rock some distance away from the cave surface. In this cavern the fuel, placed in large concrete balls, was to be stored in such a way that natural air convection would keep it cool until the repository was sealed and filled with groundwater.

After various concepts had been considered for top priority in the planned feasibility study, the variant which later became developed into the KBS-3 concept was selected. The main reason was that proof of long lasting effectiveness of the safety barriers was felt to be easier to provide and more convincing for a system that was based on natural features than for a system based on engineered features.

2.2 REPOSITORY LAYOUT

As a first step in SKB's work of evaluating the performance and safety of the WP-Cave, a detailed layout study was carried out. The design of the inner part, the storage section, was analyzed in detail and exact dimensions were given to the different shafts, drifts, distances between openings etc. The dimensions of each cave were based on a 1500 tonne storage capacity.

The central part of the repository consists of a vertical shaft with an assumed diameter of 14 m. From this shaft canister channels are directed radially outwards with an inclination downwards of 30° from the horizontal plane. These channels are 1.7 m in diameter. The canisters, with a diameter of 1.3 m, are placed centrally in the channels, so that enough space is left between the canisters and the rock wall for ventilation air to pass during the cooling period. The length of the storage section with the channels is about 16 m, which is enough to provide space for three canisters in one row. The number of channel levels is determined by the desired storage capacity. The dimensions are based on a canister capacity of 17 BWR-assemblies and a total capacity of about 1500 tonnes of uranium. This gives a total of 14 channel levels.

Due to the cooling requirements, each channel is connected to a shaft at each end for air circulation. The outer shafts are designed for cool air to flow downwards and the inner shaft for warm air to flow upwards. Before the air is recirculated it is cooled at the top. The distance between the central shaft and the inner ventilation shafts is 5 m, which is considered appropriate from a rock mechanical point of view.

The over all diameter of the inner part is 64 m measured over the outer ventilation shafts. Outside the storage space a certain volume of rock is left as a heat protecting shield for the bentonite-sand barrier. For rock mechanical reasons a thickness of about 30 m is judged to be on the safe side, thus giving a distance from the centre to the inner part of the bentonite-sand barrier of 60 m.

A width of 5 m for the bentonite-sand barrier is judged sufficient, mainly based on the space required by the high-productivity equipment used for the excavation.

Outside the bentonite-sand barrier, at a distance of 50 m, a hydraulic cage is assumed to be placed. The purpose of the cage is to short-circuit the hydraulic gradients in the vicinity of the repository. This would most probably require drill holes inclined in accordance with the fracture geometry at the site. The drains are assumed to be drilled from annular tunnels at different levels, three with a vertical distance of 145 m. The drains are 0.15 m in diameter.

Main transportation shafts and rock hoisting shafts are arranged centrally for several caves. Each cave is reached via drifts from this central area.

The depth of the cave is mainly dependent on site-specific properties. The top must be located sufficiently deep below the surface so that the hydraulic environment provides reducing conditions. The top of the hydraulic cage should provide open channels for a rapid transport of nuclides directly to the biosphere. A depth of 200 m below surface for the top of the bentonite-sand barrier was initially considered sufficient.

A deeper location would add only minor costs to the total sum and is a parameter of negligible importance from the investment point of view.

The important constraint on depth is the rock mechanics situation, since stresses increase with depth. In Sweden there is experience of mining down to a depth of 1000 m, which requires much more careful planning than mining at a depth of 500 m. However, no obvious obstacles can be seen to lowering the depth of the cave by several hundred meters.

Three modifications have since been made, which have resulted in the present reference design, the WPC 1100. First, the canisters were modified to accommodate one complete cassette from CLAB (16 BWR assemblies). Second, the Swedish nuclear program comprises about 7800 tonnes of uranium (1988 estimate), which means that five caves should have a capacity of 1560 tonnes of uranium each. Together, these two modifications resulted in the addition of two more canister channel levels, which increase the height of the cave by 11 m. Third, the requirement on temperature limitation was met by reducing the number of canisters in each channel from three to two, increasing the number of caves needed from five to seven, cf **Section 2.3.**

As a result of the adopted restriction in peak temperature, 150°C, it was judged appropriate to maintain the basic design and reduce storage capacity, which then required removing one of three canisters. By assuming that only the two outermost canister positions were used in each channel, capacity was reduced to 1100 tonnes of uranium without changing the size of the cave. This solution does not optimize the design, but this was not the aim of the analysis. Primarily it is considered essential to establish the safety potential and to define the main variables affecting safety before meaningful optimization of the design can start.

2.3 DESIGN FACTORS IMPORTANT TO THE PERFORMANCE AND SAFETY OF THE CAVE

During the studies of the different chemical and mechanical processes occurring in and around a WP-Cave after repository closure, some factors that could be of importance to the performance and safety of the repository were identified. They are also described in Section 2.5 describing the long-term performance of the WP-Cave. The factors identified that are related to the design of the repository are:

- the high temperature obtained near the canisters for hundreds of years after repository closure,
- the release through the bentonite-sand barrier of the hydrogen produced by the anaerobic corrosion of the steel canisters and the steel liners covering the shaft and channel walls,
- the chemical environment obtained in the part of the repository located inside the bentonite-sand barrier when processes such as steel corrosion, concrete degradation and weathering interact.

Because of uncertainties in thermodynamic data at higher temperatures, the maximum temperature allowed in the repository was set to 150°C. Temperature calculations have shown that this limit will be exceeded in the 1500 tonne capacity design with shafts and channels filled with ground rock material. The high temperature is partly a result of the finely ground rock backfill, which will maintain almost stagnant water in the shafts and channels. In spite of the negative effect on the temperature, the alternative with ground rock, in the shafts and channels was selected for further investigation. This is motivated by the positive effect of the ground rock material in terms of increased sorption capacity and slowly moving or stagnant water, which will result in lower transport rates of dissolved nuclides through the channels and shafts. The ground rock will also help keep the corrosion products and the fuel in place when the canisters are degraded by corrosion. A finely ground rock material might also act as a filter for colloids.

To meet the temperature limit of 150°C it was decided to reduce the number of canisters in each canister channel from three to two. This means that the amount of fuel to be stored in one 1500-tonne-capacity WP-Cave is reduced to 1100 tonnes. The dimensions of the repository are kept unchanged.

The anaerobic corrosion of the steel material in the repository will produce hydrogen. The gas will probably flow to the upper part of the repository and form a bubble inside the top of the bentonite-sand barrier. If not released, the bubble could displace water from the inner parts of the cave. The escape of gas through the bentonite-sand barrier will start when the pressure inside is larger than the critical pressure for the barrier. The critical pressure for gas flow through a bentonite-sand mixture increases with greater amounts of bentonite in the mixture. It has been found that a mixture with 10% bentonite will result in a sufficiently high gas penetration rate through the top part of the barrier. In the safety analysis a reduction of the amount of bentonite in the top part of the barrier from the original 50% down to 10% was therefore made.

Having a large amount of concrete present in the repository will increase the difficulties in defining the chemical environment, since concrete degradation at temperatures up to 150°C has to be considered in addition to anaerobic steel corrosion and weathering processes. It was therefore decided that the further investigation should be devoted to an alternative where the amount of concrete present in the repository is so small that the effect on the chemical environment will be negligible.

The revision of the original 1500 tonne capacity design that was made, based on the long term chemical and mechanical processes occurring in a WP-Cave repository, can then be summarized as follows:

- the number of canisters in each canister channel was reduced from three to two, giving a reduction in the total amount of fuel in the repository to about 1100 tonnes,
- the percentage of bentonite in the top part of the bentonite-sand barrier was reduced to 10%,
- the amount of concrete present in the facility was to be kept so small that its effect on the chemical environment is negligible.

2.4 COST ESTIMATES

2.4.1 General

A total back-end cost analysis of the WP-Cave concept was conducted at the 1986 price level, whereby the differences in comparison with KBS-3 were identified and the costs of the WP-Cave used instead.

2.4.2 Total Back-end Costs with the WPC 1100

The total future cost from 1987 and onwards of the Swedish back-end system, assuming a final repository for high-level waste according to the WPC 1100 concept (seven caves), has been calculated to be MSEK 55 600 at the price level in January 1986, cf Table 2-1. This is MSEK 16 300 or 42% higher than the estimated cost of the KBS-3 concept, MSEK 39 300 in PLAN 86 /2-1/, which was used as a basis for the calculation of the required back-end fee on nuclear electricity production. When considering only handling and final disposal of the spent fuel the costs are MSEK 44 300 for the WPC 1100 and MSEK 28 000 for KBS-3.

The major differences between the two concepts are a cost increase of MSEK 20 800 for the WP-Cave underground repository, including supervisional costs during the long cooling and supervision period, an estimated decrease of the cost of encapsulation by MSEK 2 800 and a decrease in the operating costs of CLAB by MSEK 1 400.

The lower encapsulation costs for steel canisters are a result of excluding the lead casting and the cooling, as well as a decrease in the time needed for encapsulation of all the spent fuel. The cost of construction material — copper or steel — give very small differences in the total sum.

For CLAB, the decrease is the result of the shorter encapsulation time leading to an earlier emptying of the pools and thus a shorter operating period for CLAB.

Table 2-1. Comparison of total future cost of the Swedish back-end system including seven WPC 1100 or one KBS-3. Costs (MSEK) from 1987 in the price level of January 1986.

Object	Cost WPC	Cost KBS	Cost difference + denotes higher for WPC
SKB-adm and R&D	3 158	3 158	0
Transport	1 316	1 509	-193
CLAB	4 956	6 331	-1 375
Final repository			
- Common facilities	3 525	3 531	-6
- Encapsulation station	4 038	6 866	-2 828
- Repository for spent fuel SFL 2	26 008	5 216	20 792
- Repository for other high-level waste SFL 3–5	1 336	1 395	-59
SUBTOTAL handling and final disposal of spent fuel	**44 337**	**28 006**	**+ 16.331**
Decomm. of NPP	7 563	7 563	0
Final repository for operational and NPP decomm. waste SFR 1-3	1 143	1 143	0
Reprocessing	2 614	2 614	0
TOTAL future costs of the Swedish back-end system	**55 657**	**39 326**	**+ 16 331**

2.4.3 Major Factors Influencing the Total Costs

Four factors have a major impact on the total costs for the underground part of the repository:

– the size of the WP-Cave, that is its storage capacity,
– the width of the bentonite-sand barrier,
– the quality of the bentonite,
– the quality of the sand.

These factors are highlighted below.

WPC 3750

The cost analysis for this size assumes a repository that is large enough for storing 3750 tonnes. The design dimensions do not take into account the peak temperature limit adopted in the present safety analysis. The difference in **construction** costs are shown in Table 2-2.

Table 2-2. Comparison of construction costs for different sizes, different bentonite-sand barrier thicknesses and different bentonite and sand qualities in backfill material, final repository for high-level waste, SFL 2. Costs in MSEK at the price level in January 1986.

The following abbreviations are used for the different qualities of bentonite and sand:

WB = Wyoming bentonite
IB = Italian bentonite
BS = quartz sand from Bornholm
BB = ballast made from hoisted rock

WPC 1100 (seven caves)

Thickness of bentonite-sand barrier	5 m			2.5 m		
Bentonite	WB	WB	IB	WB	WB	IB
Sand	BS	BB	BB	BS	BB	BB
Costs	26 000	21 600	19 800	18 300	16 000	15 000

WPC 3750 (two caves)

Thickness of bentonite-sand barrier	5 m			2.5 m		
Bentonite	WB	WB	IB	WB	WB	IB
Sand	BS	BB	BB	BS	BB	BB
Costs	15 700	13 100	11 800	10 700	9 300	8 700

Bentonite-sand Barrier Thickness

A barrier thickness of 5 m was originally chosen as this width in mining is known to provide sufficient space for high productivity equipment in rock excavation and slot backfilling. A decrease of the thickness to 2.5 m is judged possible from a construction point of view. The economical outcome of such a change has been estimated as presented in Table 2-2.

Bentonite Quality

The cost estimate is based on an assumed use of high quality sodium bentonite, denoted as Wyoming bentonite. In the SFR facility, however, an activated calcium-rich bentonite from Italy was used. This product is less expensive today than the Wyoming product.

The sand material considered is a quartz-sand with good thermal conductivity. The price of the high-quality product, of metallurgical grade, is high in Sweden. The main source is the island of Bornholm. A low cost material would be a concrete ballast material, crushed and screened in a plant at the site of the final repository.

The differences in construction costs assuming use of the different qualities are shown in Table 2-2.

2.5 RESULTS OF THE PERFORMANCE STUDIES

2.5.1 Analysis Strategy

The performance and safety analysis that has been carried out within the SKB-WP-Cave project was performed in three separate phases. In the first phase a description of the expected conditions in and around the repository during construction, short term and long term storage was made. The primary aim was to identify areas where more detailed studies were required before a performance assessment of the concept could be made.

The second phase was devoted to studies of some of the areas identified in part one. Since the big difference between the WP-Cave concept and the KBS-3 concept is to be found in the near field of the repository, the analysis was concentrated on processes occurring here, such as:

- the composition of the water in the part of the repository located inside the bentonite-sand barrier,
- the effect of radiolysis of the water in contact with the fuel,
- some preliminary calculations of uranium and plutonium solubility,
- the transport of gas and water through the bentonite-sand barrier,
- the temperature distribution when shafts and channels are backfilled with sand and water,
- the mechanical effects on the rock mass inside the bentonite-sand barrier caused by:
  - the excavation of the slot for the bentonite-sand barrier
  - the heat generated by the fuel
  - the increase in volume of the canister material during corrosion,
- the thermally induced convective movement of groundwater in the near field.

Some initial calculations of the radionuclide transport in the near field barriers and of the nuclide migration in the geosphere (far field) were also performed. The primary aim of these calculations was to gain a better understanding of the behavior of the system, with a special focus on the transport processes, and to identify the relative importance of the different barriers for radionuclide transport to the biosphere.

In the third phase, radionuclide transport from the fuel through the near field, the far field and the biosphere was studied. The primary aim was to get an indication of the magnitude of the individual doses that could be expected from the radionuclides released from a WP-Cave repository. In setting up the scenarios, the results obtained from the investigations conducted in phase two were taken into account. It was also necessary to make a number of assumptions and simplifications in defining the scenarios and in modeling of the nuclide transport. In general, a conservative approach was used. In some cases, however, data availability and/or the present level of knowledge were the governing factors. The scenarios with assumptions and simplifications are described in the following Subsections.

The groundwater flow rate through the repository was found to be very sensitive to assumptions regarding the conductivities of the clay barrier and the repository interiors, particularly the former.

Consequently, a case with low conductivities, referred to as the "low-flow-through" case, and a case with high conductivities, referred to as the "high-flow-through" case, have been evaluated.

The performance assessment is presented in more detail in a separate report /2-2/.

2.5.2 Low-flow-through Case

A summary of the analysis of the low-flow-through case for radionuclide transport through the near field, the far field and the biosphere is presented in the following. The premises of the low-flow-through case are based on the results of the previous mentioned studies of different processes occurring in and around a WP-Cave, and on experience gained during the NAK-WP-Cave project and studies of other repository concepts such as KBS-3 and SFR. The premises as well as the modeling and the results for the near field, the far field and biosphere transport are presented.

Near Field

In the near field analysis, nuclide migration from the fuel in the canisters to the water flowing in the rock outside the bentonite-sand barrier was calculated.

During a time period of 100 years after the canisters are placed in the canister channels, the repository is kept dry. The heat generated by the fuel is removed by air circulating in the ventilation shafts. After the dry period, the channels and the shafts are backfilled with finely ground rock material and water, and the repository is sealed off.

The water in the repository is initially saturated with oxygen which, however, is rapidly consumed by oxidative corrosion of the iron present in the repository. After that, anaerobic corrosion of the mild steel canisters and the steel liners covering the channel and shaft walls occurs.

Due to the high temperature maintained in the repository for several hundred years after closure, anaerobic corrosion of the canisters may proceed relatively fast. It was assumed here that water penetration occurs in all canisters 200 years after repository closure. The fuel is then 340 years old, as a result of the 40 year wet storage period in the CLAB facility and the 100 year dry storage period.

Anaerobic corrosion generates divalent iron corrosion products, and it can therefore be assumed that reducing conditions prevail everywhere from the corroded canister walls to the rock outside the bentonite-sand barrier even after all iron metal has corroded away. The corrosion process also generates large amounts of gaseous hydrogen. For the purpose of radionuclide migration calculations, the gas was assumed to escape at the top of the cave, causing no interference with the migratory process.

The fuel matrix dissolution model used was somewhat different from the one used in the KBS-3 study. The analysis for the WP-Cave assumes that the fuel is oxidized directly by oxidants formed by radiolysis. The best estimate was a 1.2 million year leach time with an initial $6 \cdot 10^{-5}$ per year oxidation rate. The fuel-to-clad gap and

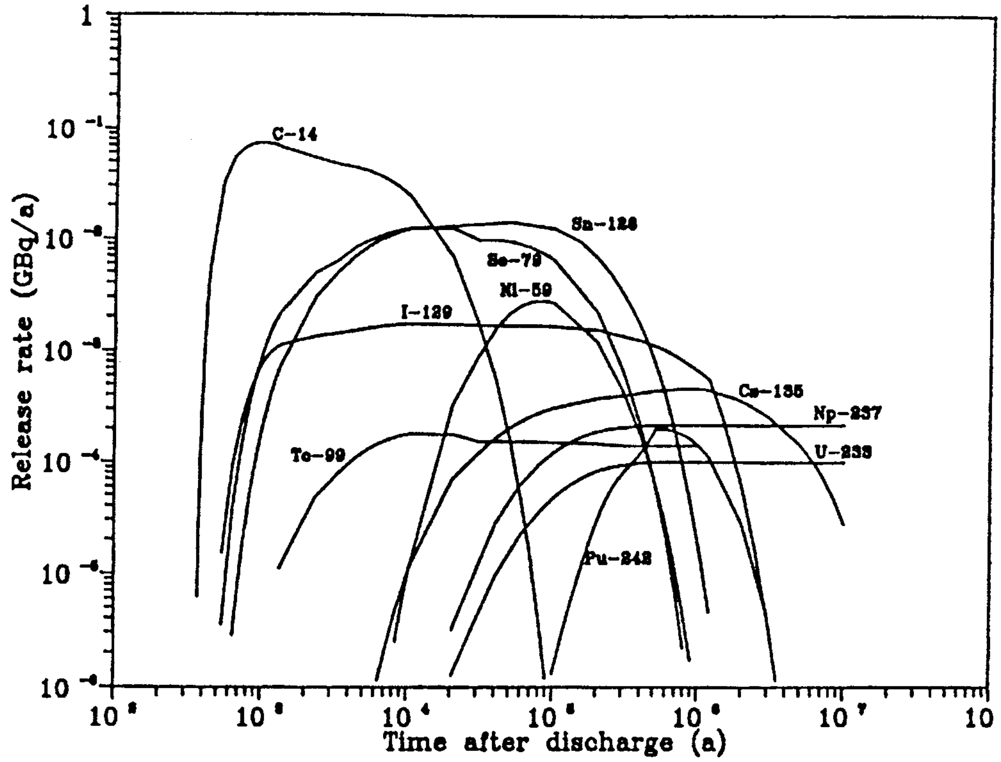


Figure 2-1. Release rates of the most important nuclides from the bentonite-sand barrier.

grain boundary inventory of carbon-14, iodine and cesium was assumed to escape immediately after canister penetration.

The processes assumed to control the transport of radionuclides from the canister to the outer surface of the bentonite-sand barrier were

- Diffusion and sorption in the crushed rock used as backfill,
- Diffusion and sorption in the bentonite-sand barrier,
- Diffusion in the nearly stagnant water immediately outside the bentonite-sand barrier,
- Advection due to thermally induced flow through the top of the bentonite-sand barrier during the first 3000 years after canister failure.

Early computations indicated that possible sorption in the rock between the backfill and the clay was of very little importance.

Calculations involving unlimited solubilities were also made for comparison but are not presented here.

The calculated release rates of the dominant nuclides from the bentonite-sand barrier near field are shown in Figure 2-1.

Far Field

The release to the biosphere of radionuclides escaping from the bentonite-sand barrier was calculated in the far field analysis. The hydraulic cage was not taken into account in the computer calculations; the effects of the cage are treated separately below.

The radionuclides released from the bentonite-sand barrier will be transported by the water flowing in fractures in the rock surrounding the repository. Nuclides will sorb on those parts of the fracture surfaces that are in contact with the nuclide contaminated water. Diffusion of nuclides into the stagnant water in microfissures and pores connected to the wetted fracture surfaces and sorption on the inner surfaces of the rock matrix will also occur.

One conservative interpretation of observations made in Stripa, SFR and two tunnel works in Sweden is that water flow is very unevenly distributed in fractured crystalline rock. Based on this interpretation, it has been suggested that water flow be described as flow in "channels" in the fractured rock. The channels are characterised by a certain width and a certain flow rate. Different channels have different flow rates, and the variation in flow rate is considerable. To determine the radionuclide migration in the rock it is therefore necessary to know the distribution of flow between the different channels as well as their wetted surface, the length of the channels, the sorption properties of the rock minerals, the porosity of the rock matrix and the nuclide diffusivity in the rock matrix.

A transport model based on the channeling concept was used to calculate the release to the biosphere of nuclides escaping from the bentonite-sand barrier. In the model the rock is described as a bundle of independent channels each with its own flow rate and wetted surface. The transport in each channel is calculated individually and the sum of the effluents is calculated by adding together the effluent concentration of all channels. Hydrodynamic dispersion in the channels is neglected because "dispersion" due to the flow rate distribution will be considerably larger than that due to hydrodynamic dispersion. Sorption and diffusion into the rock matrix is, however, considered.

It was assumed that the flow rate distribution observed in the four main tunnels in the SFR repository could be extended to the rock mass between the WP-Cave repository and the biosphere. Data on channel frequency and the fraction of the total flow in each channel group were taken from the evaluation of the observations made at the SFR repository site.

The results of the calculations show that very little retardation of the fission and activation products occurs in the rock over a travel distance of 100 meters and a channel width of 1 meter.

The reduction in release caused by sorption and decay during the far field transport is less than a factor 10 for parent and early daughter nuclides in the decay chains. The release of late daughter nuclides will be somewhat higher from the far field compared to the near field. This is due to the production of late daughters that occurs during the far field transport.

If oxidizing instead of reducing conditions prevail in the far field, less sorption of technetium, uranium and neptunium will occur. The effect of the far field for these nuclides and their daughters is in this case negligible.

In all, the maximum release rate from the far field is, in general, about 50% of the maximum release rate from the near field. This value is somewhat smaller for nuclides with short half lives. For long-lived nuclides, the near field and far field release is almost equal.

The channeling model that has been used here for radionuclide transport in the far field is different from the advection-dispersion model based on porous media flow that was used in the KBS-3 study. The difference lies mainly in the amount of wetted surface and, hence, the amount of rock available for sorption, which is important for retardation in the geosphere. Another difference is the limitation of the main flow to a few noninteracting channels. It is probable that mixing occurs, at least over long travel distances. The fast channels will in that case not dominate the transport, and considerably lower release rates from the far field will be obtained.

The Hydraulic Cage

In the low-flow-through case calculations presented in the previous section, the presence of a hydraulic cage was not considered. The possible effects of the hydraulic cage were, however, estimated separately /2-2/.

Biosphere

The turnover in the biosphere of radionuclides released from the far field was assessed in the biosphere analysis. The ecosystem was chosen to be the same as the one previously used in the analysis of the KBS-3 concept. The area represents a typical landscape in the middle of Sweden consisting mostly of woodlands containing small agricultural areas, a well and a lake.

The well is assumed to be located in the repository discharge area. The nuclide concentration in the well is dependent on the ratio of nuclide-contaminated water to non-nuclide-contaminated water reaching the well. At present it is not clear how this ratio should be estimated. Because of this, two different values of dilution were assumed. One is the same as that used in the KBS-3 study, where it was assumed that the nuclide concentration in the water in the well corresponds to the nuclide release from the far field each year diluted in 500 000 m^3 water. Because of differences in design and depth of location between a KBS-3 repository and a WP-Cave, the dilution will most probably be less than in a well near a WP-Cave. It was therefore also assumed that the annual nuclide release from the far field is diluted in 2 000 m^3 of water. This dilution is based on the estimated extraction rate and single-family consumption rate ($\approx$6 m^3/day). The dilution must be regarded as extremely small. The possibility of this extreme scenario occurring cannot be ruled out entirely, however.

The compartment model BIOPATH was used to calculate the nuclide transport in the biosphere.

Resulting Dose to Man

In Table 2-3 a comparison between dilution in 500 000 m^3 per year and dilution in 2 000 m^3 per year is made for some of the dominant nuclides. With the large dilution, the maximum total dose in the combined well and lake scenario is determined by the fission and activation products and is less than 10 μSv per year.

With the small dilution, the maximum total dose obtained both in the well and in the combined well and lake scenario is in the order of 100 to 200 μSv per year. The dose obtained from actinides and daughters is of the same magnitude as the dose obtained from fission and activation products.

Table 2-3. **<u>Low-flow-through</u> results. Maximum annual doses obtained in the combined well and lake scenario from the dominant nuclides. Comparison between large and small dilution.**

Nuclide	Maximum dose, (Sv/a), dilution in 500 000 m^3 per year well/lake	2 000 m^3 per year well/lake
^{14}C	$6.2 \cdot 10^{-6}$	$1.5 \cdot 10^{-5}$
^{79}Se	$7.2 \cdot 10^{-7}$	$9.7 \cdot 10^{-6}$
^{126}Sn	$4.2 \cdot 10^{-7}$	$1.6 \cdot 10^{-5}$
^{129}I	$8.5 \cdot 10^{-7}$	$6.7 \cdot 10^{-5}$
^{237}Np	$2.6 \cdot 10^{-7}$	$5.9 \cdot 10^{-5}$
^{229}Th	$1.5 \cdot 10^{-7}$	$3.1 \cdot 10^{-5}$
^{242}Pu	$5.8 \cdot 10^{-8}$	$1.1 \cdot 10^{-5}$
^{226}Ra	$1.0 \cdot 10^{-9}$	$2.3 \cdot 10^{-7}$
^{210}Pb	$1.5 \cdot 10^{-8}$	$9.0 \cdot 10^{-7}$
^{231}Pa	$3.1 \cdot 10^{-9}$	$6.6 \cdot 10^{-7}$
Fission and activation prod. (first peak)	$8.2 \cdot 10^{-6}$	$1.1 \cdot 10^{-4}$
Actinides and daughters (second peak)	$4.9 \cdot 10^{-7}$	$1.0 \cdot 10^{-4}$

2.5.3 High-flow-through Case

With the hydraulic conductivities in the bentonite-sand barrier and in the interior rock mass assumed in the low-flow-through case, 10^{-11} and 10^{-9} m/s respectively, the release of nuclides from the bentonite-sand barrier is dominated by diffusion except for the first 3000 years after canister breakthrough when nuclide release from the top of the barrier by thermally induced flow dominates. Experimentally measured hydraulic conductivities in 10/90 bentonite-sand mixtures indicate, however, the possibility of 10 times higher values than assumed in the low-flow-through case, i.e., 10^{-10} m/s /2-3/. The excavation of the slot for the bentonite-sand barrier will cause an opening of horizontal and vertical fracture structures in the interior rock mass which probably will result in considerable higher hydraulic conductivity in that part of the rock than that assumed in the low-flow-through case /2-2/.

With a hydraulic conductivity of 10^{-10} m/s in the bentonite-sand barrier and 10^{-7} m/s in the interior rock mass, the thermally induced flow through the top of the bentonite-sand barrier will be approximately twenty times higher than in the low-flow-through case. Another effect of these higher conductivities is an increase in the natural-gradient-induced water flow through the bentonite-sand barrier resulting in a nuclide release from the barrier that is determined by flow over longer periods as well, and not, as in the reference case, by diffusion.

To study the effect of higher hydraulic conductivities in the bentonite-sand barrier and in the interior rock mass, a high-flow-through case for near field transport was set up. The only difference from the low-flow-through case is the above-mentioned higher thermally induced flow by which the nuclides are released from the top of the bentonite-sand barrier at early times, and the natural-gradient-induced flow by which the nuclides are released from the top of the barrier when the thermally induced flow has declined.

The maximum release rates from the near field for this case are given in Table 2-4 for the important fission and activation products and for the mother nuclides in the decay chains. The higher hydraulic conductivities in the bentonite-sand barrier and in the interior rock mass result in considerably higher release rates from the bentonite-sand barrier compared to the low-flow-through results. Consequently, the doses to man will also be higher. For a combined well and lake scenario with a dilution in the well of 500 000 m^3 per year, the maximum dose obtained from only carbon-14 is 200 μSv per year. With the small dilution in the well of 2 000 m^3 per year the total maximum dose will be approximately 2 mSv per year.

Table 2-4. High-flow-through results. Maximum annual dose for the combined well and lake scenario.

Nuclide	Maximum dose, (Sv/a), dilution in 500 000 $m^3 \cdot a^{-1}$ well/lake	2 000 $m^3 \cdot a^{-1}$ well/lake
^{14}C	$2.0 \cdot 10^{-4}$	$4.8 \cdot 10^{-4}$
^{79}Se	$4.3 \cdot 10^{-6}$	$5.8 \cdot 10^{-5}$
^{126}Sn	$2.9 \cdot 10^{-6}$	$1.1 \cdot 10^{-4}$
^{129}I	$1.6 \cdot 10^{-5}$	$1.3 \cdot 10^{-3}$
237Np	$1.6 \cdot 10^{-6}$	$3.5 \cdot 10^{-4}$
^{229}Th	$9.0 \cdot 10^{-7}$	$1.9 \cdot 10^{-4}$
^{242}Pu	$6.4 \cdot 10^{-7}$	$1.2 \cdot 10^{-4}$
Fission and activation prod. (first peak)	$2.2 \cdot 10^{-4}$	$2.0 \cdot 10^{-3}$
Actinides and daughters (second peak)	$3.1 \cdot 10^{-6}$	$6.6 \cdot 10^{-4}$

2.5.4 Other Important Performance and Safety Factors

The performance assessment also involved estimates of the sensitivity of the results with respect to some of the assumptions made in the low-flow-through case. In addition, a more qualitative analysis was made of some assumptions that were not quantitatively estimated.

A few major influential factors were identified. These were:

- the flow rates through and around the bentonite-sand barrier,
- sorption data and solubilities at temperatures up to 100—150°C,
- the flow situation in the geosphere in terms of wetted fracture surface, flow distribution and transport length, and dilution of the contaminants in a well.

2.5.5 Assessment of Alternative Design Features

Thickness of the Bentonite-sand Barrier

The thickness of the bentonite-sand barrier in the low-flow-through case is 5 m. In order to investigate the sensitivity of near field release to barrier thickness, the release rates from a 2.5 m thick bentonite-sand barrier were estimated.

It was found that the maximum release rate of the dominant nuclides was less than five times higher from a 2.5 m thick bentonite-sand barrier than from a 5 m thick barrier in the low-flow-through case, and about a factor of two higher in the high-flow-through case. Release rates several orders of magnitude higher from a 2.5 m compared to a 5 m thick barrier were found for some radionuclides with high sorption capability in the barrier and short half lives. The contribution made by these nuclides to the total dose from these nuclides is, however, negligible in the case of a 5 m thick barrier, and will be very small in the case of a 2.5 m thick barrier.

2.6 CONCLUSIONS FROM THE COST AND PERFORMANCE ANALYSIS

Based on the original design of WP-Cave, a repository with a capacity of up to about 1 100 tonnes of U and a bentonite-sand barrier thickness of 5 m filled with Wyoming bentonite and quartz-sand represents a higher cost than the KBS-3 concept, which is considered in the cost calculations for the Swedish back-end system. The difference in future costs (from 1987) for the total Swedish back-end system is estimated to about MSEK 16 300 when comparison is made between a repository with seven WPC 1100s and a KBS-3 repository. Cost figures are given at the price level of January 1986.

The potential for reducing the costs is associated with three parameters:

- The capacity of each cave,
- The thickness of the bentonite-sand barrier,
- The quality requirements on the bentonite and the sand material.

Particular measures must be taken in order to lower the costs of the WP-Cave concept to the same level as those of the KBS-3 concept. Choosing less expensive bentonite and ballast grades has not been considered in the performance and safety analysis. The question of concern so far has been the requirement on a very low hydraulic conductivity, good swelling properties and good sorption coefficients. The most important factor is hydraulic conductivity.

The thickness of the barrier may be difficult to reduce from 5 m to 2.5 m.

The design with a higher capacity has not been adapted to the peak temperature limitation, which will reduce the capacity of the assumed design.

Any cost comparison should also take note of the fact that cost saving possibilities exist for the KBS-3 concept as well. One example is the grades of bentonite and sand, which are assumed to be the expensive grades today.

Further options for cost-effective design are:

- No bottom cone in the bentonite-sand barrier,
- Increased height of the cave and thus increased capacity with the same thermal load per cubic meter of rock,
- Optimal design of the storage space with regard to excavation methods.

It therefore can be concluded that the design of WP-Cave has not been optimized from an economic point of view.

The cost evaluations performed so far in the WP-Cave analysis indicate that the basic design of WP-Cave currently being considered, WPC 1100, would add a substantial cost to the total costs of the Swedish back-end system.

The results from the performance and safety assessment of the proposed WPC 1100 show that radionuclides could be released to the biosphere in such concentrations that established radiation protection criteria would be violated. The most conservative scenarios analyzed include a number of unfavorable assumptions regarding drilled wells, fast radionuclide transport channels, groundwater flow through the repository and radionuclide solubilities. Further refinement of the proposed design is needed if the concept is to be pursued.

The temperature peak is of concern from three aspects: sorption data, solubilities and thermally induced water flow. The same sorption data and solubilities are used as in the KBS-3 study. They have been determined at room temperature and will most probably change with increasing temperature. Higher sorption of the short-lived dose dominating fission and activation products, especially in the bentonite-sand barrier, would reduce the total exposure to man. The resulting effect of changes in solubilities cannot be firmly quantified at present.

The thermally induced flow dominates the release from the near field during the initial period. The gradient is dependent on the thermal load in the WP-Cave. A radical decrease in load is not considered possible from an economic point of view. The flow resistance is composed of the resistance in the bentonite-sand barrier and in the inner rock mass. In the high-flow-through case, the bentonite-sand barrier has been assumed to have a hydraulic conductivity of 10^{-10} m/s and the inner rock mass about 10^{-7} m/s. The flow resistance is then dominated by the bentonite-sand barrier. If the hydraulic conductivity of this barrier were reduced by a factor of 10, the flow and subsequently the release rate would also be reduced by a factor of 10. This also affects the maximum release rate by a factor of 10. The hydraulic conductivity cannot be reduced much in the bottom part with the present barrier design, but it can in the top part if the migration of hydrogen gas permits.

An improvement of the resistance of the inner part is not judged to be feasible. The conductivity must be reduced by at least two orders of magnitude, which seems unrealistic for this fractured large volume of rock.

Other options considered to explore the potential of improvements in the design, are a new design of the bottom cone, so that pure bentonite can be used, and the introduction of an impermeable layer in the bentonite-sand barrier that also covers the cylindrical part of the barrier. In the case where pure bentonite is used, the inner rock mass must be supported by rock pillars, in the same way as in the very early design of WP-Cave. If any of these options is adopted it must be noted that the safety of the repository is made very much dependent on one barrier only.

In the long run, when the temperature has dropped to ambient, natural flow dominates over diffusion as a mechanism for nuclide releases to the flowing water outside the bentonite-sand barrier in the high-flow-through case. This is when no hydraulic cage is considered. If a cage operates with an efficiency of 90%, the flow releases are reduced to nuclide quantities of the same order of magnitude as those released by diffusion. The efficiency of the cage might possibly be improved somewhat. Other measures to reduce the importance of natural flow are the improvements in hydraulic conductivity of the bentonite-sand barrier discussed above.

From the above it should be obvious that efforts would have to be made to improve the barrier properties of the bentonite-sand slot. Thus it is less probable that measures will be judged feasible that are meant to simplify construction in order to reduce costs, e.g. reducing the barrier thickness and introducing a less expensive but not so smectite-rich bentonite quality. The exchange of quartz sand for crushed rock, however, seems to have no impact on the quality of the bentonite-sand barrier.

The hydraulic cage is part of the barrier system. The cage, when functioning, is expected to reduce the flow through the bentonite-sand barrier and thus the nuclide release by flow. It is further expected to reduce the water velocity along the outer surface of the bentonite-sand barrier, which affects the nuclide release by diffusion and/or short-circuits parts of the ground water channel system, which limits the water flow past the bentonite-sand barrier. On the other hand the cage, when successfully constructed, intersects existing fracture systems with connections to the surface. Due to this it may be considered more feasible to develop the small well dilution case. No negative impact has been noticed from the cage in the study of thermally induced flow. With the model and data assumptions made in the calculations, the hydraulic cage has positive barrier functions.

Some major options for lowering the costs of the WP-Cave have been identified such as: larger size of cave, thinner bentonite-sand barrier and less expensive bentonite and sand grades. In view of the conclusions presented here, only the quartz sand option seems to be feasible at present. The other options run contrary the general aim of improving the safety of the present design.

An option not explored here is the possibility of substituting copper for steel as the canister material. Such a solution would seem to improve the safety of a WP-Cave concept and neutralize some negative impacts of the high temperature, but it would also increase the costs somewhat.

3 ASSESSMENT OF THE POTENTIAL FOR SAFETY AND DEVELOPMENT OF THE WP-CAVE

3.1 INTRODUCTION

WP-Cave is discussed and compared with the reference concept of KBS-3 on the basis of a performance assessment made in 1987/88 and described in /3-1/ and the KBS-3 report from 1983 /3-2/.

Some of the differences stem from the selection of materials or dimensions for the barriers. When distinctive differences are found a brief discussion is often held in the text concerning the possibility of transferring the beneficial aspects to the other concept or of changing a problematic design. Such changes are not analyzed in detail, they are only pointed at to indicate where flexibility of design exists.

Other differences are closely linked to the basic aim of the WP-Cave, to pack the spent fuel densely in a small repository volume that could be sited with only limited requirements on the quality of the host rock. Examples of such features are the 100 year period of active cooling required, and the higher temperatures in the near field of the waste canisters. These aspects are also discussed on a more theoretical level with regard to their effect on the provability and level of confidence of the safety evaluation, and on the possibility of developing the concept into a licensed facility.

3.2 THE PERIOD OF ACTIVE COOLING OF THE WP-CAVE

Disposal of large quantities of spent fuel in a small repository volume gives rise to high temperatures. The WP-Cave is based on two temperature design criteria:

- The temperature in the bentonite-sand barrier shall not exceed 80°C,
- The maximum temperature on the canister's surface shall not exceed 150°C.

These criteria are met by selecting a proper size for the cave and by limiting heat generation at the time of the final sealing of the repository. The decay heat declines slowly, and for the WP-Cave it is necessary to cool the fuel for at least 100 years before abandoning the repository. The repository design includes an air ventilation system that can be operated for the time required.

The design further allows for an extension of the ventilation period by another 100 years, if this is found desirable. As long as an active cooling system is installed, the cooling capacity is a question of how the heat exchange system is designed. Thus the fuel can be stored in the WP-Cave very soon after it has been discharged from the reactor. In this respect the repository can be regarded as an interim storage facility, which is later converted into a final repository.

With the given layout, expected burn-up of the fuel and temperature criteria, the repository will be kept open until the spent fuel is 140 years old: 40 years of interim storage in CLAB and 100 years of storage in the WP-Cave. This time period cannot be shortened without adjustments in design or storage capacity. For KBS-3, the final sealing of the tunnel and shaft system is planned to take place around the year 2050, about 40 years after the last fuel was taken out of the reactors, although it is possible to postpone this date.

The required supervision over a long time period will, however, also provide an opportunity for long-term monitoring of certain functions of the repository, such as:

– Heat generation compared to predictions,
– Heat transport in the near field, bentonite-sand barrier and the far field,
– Water uptake by the bentonite,
– Groundwater flow through the hydraulic cage,
– Hydrogeology and hydrochemistry in the far field,
– Hydrogeological interaction between neighboring areas hosting one WP-Cave each.

Based on present-day knowledge, however, the effect of temperatures on safety seem to be more closely connected with the sealed repository and a time span around 1 000 years and longer than with an open repository and a period of 100 years. The higher level of confidence in the safety assessments gained in a hundred-year measurement program is thus doubtful.

If there is substantial doubt as to the safety of the repository and it is believed that that doubt can be cleared up with a 100 year measurement program, it would be wiser to prolong the storage period in CLAB, and delay the construction of the repository and the encapsulation until the information is available.

When WP-Cave is used for interim storage, the spent fuel will be placed deeper down in the bedrock than when it is stored in CLAB. Measures can further be taken for a self-sealing operation to be activated in case of a national disaster (earthquake, major flood or war) leading to an immediate abandoning of the facility. Such measures have, however, not been discussed in this report. The question of how skilled personnel are to be recruited is essential for the analysis of the practical feasibility of the concept, especially considering the current plans in Sweden for a phase-out of nuclear power production by 2010. The encapsulation and emplacement of the spent fuel will in the WP-Cave concept be intensive and take about 13 years. Thereafter, CLAB may be decommissioned. The only activity left concerning nuclear material is the supervision and monitoring of the WP-Cave.

The long cooling period adds a certain extra cost, but the amount is of minor importance in comparison with the total costs of the Swedish back-end system, especially when the costs are discounted to present value. The staff is estimated to number about 10 persons. The costs at 1986 price levels are estimated to total about MSEK 17 per year.

Comparing the two conceptual approaches, the WP-Cave can be said to require or provide a 100 year long dry storage period between the CLAB and the final sealed storage phase. During that phase the WP-Cave offers a higher level of retrievability when compared to a sealed KBS-3 system. A further extension of the non-sealed phase for the WP-Cave is also possible. For the KBS-3 system, a long period of supervision is not required. A basic retrievability exists as long as the waste canisters have not been penetrated, since it is always possible to remove the sealing and backfill materials in the emplacement areas. Should an easier retrievability be desired, it is best achieved by extending the storage period in CLAB. Such an extension was investigated in /3-3/ and found achievable at least up to a storage time of 100 years. The limitation was due to the lack of experience of extended storage of zircaloy-clad fuel in water pools.

A period with an intermediate level of retrievability can be provided for the KBS-3 system by delaying the backfilling of deposition tunnels. The consequences of such a strategy have not been fully investigated.

Conclusions

For a long period of high retrievability the WP-Cave system is expected to give a somewhat higher level of safety against acts of war etc than CLAB. It is, however, considered to be a drawback that the WP-Cave will always require the 100 year period of active cooling. A 100 year period for measurements in or around the repository is not expected to provide much additional information relevant to safe disposal by either the WP-Cave system or the KBS-3 system.

Since the CLAB facility is already in operation, it is felt that a high level of retrievability can easiest be achieved by an extended CLAB storage. This option should be used if reasonable doubts are raised against the safety or economy of a 2020 transfer of the fuel to encapsulation and final storage, whatever the concept.

In summary, the 100 year operating period of the WP-Cave is not considered necessary in the Swedish system.

3.3 ADAPTABILITY TO THE GEOLOGICAL ENVIRONMENT

The site investigation studies performed in Sweden have so far been designed to make it easier to detect vertical or steeply dipping fissured zones. Gently dipping fracture zones have been identified at most of the sites, however. Recent developments in seismic investigation technology, borehole radar and hydraulic interference tests have increased our ability to detect such zones and to map their extent.

A compilation of available data from different sites in Sweden has indicated that the presence of gently dipping zones is very probable at most of the studied sites.

The compilation covered about ten sites investigated by SKB in Sweden, the Siljan deep hole and two sites abroad. At eight of the Swedish sites, zones were identified with a width of more than 25 m and a hydraulic contrast of 10^2 in relation to the surrounding bedrock, or a width of 10 m and a hydraulic contrast of 10^3. Such zones could increase the radionuclide release from a WP-Cave by a factor of two or more. The information available covers a depth of about 800 m, except for the Siljan hole which is much deeper. The Siljan hole as well as the two sites abroad have several fracture zones.

The material so far indicates a high probability of fracture zones intersecting a WP-Cave with a significant impact on the release of radionuclides. Most of the investigated sites have one or several fracture zones, limiting the possibility of locating one cave or a farm of caves.

The effect of a fracture zone is largely dependent on the hydrological characteristics of the fracture. A plausible scenario is that the zone emerges in a topographically low area, which means that the hydraulic head is low in the zone compared to the head in the surrounding rock and that the zone will act as a drain. Around the WP-Cave the zone intersects the hydraulic cage, which is a large reservoir of water. Under normal circumstances the head in the cage would be somewhat higher than in the fracture zone and the zone can be assumed to drain the cage.

If a well is drilled in that zone the water can contain a large proportion of the water from the hydraulic cage. When the canisters in the cave have corroded all the radionuclides released from the cave will be moving towards the hydraulic cage. The dilution is limited mainly to infiltrating surface water.

An additional negative consequence of a highly permeable zone intersecting the hydraulic cage has been displayed in the analysis of thermally induced water flow. One scenario, discussed in Subsection 3.5.3, assumes the existence of a zone just below the center of the repository. For this scenario the vertical flow component along the outside of the bentonite-sand barrier is higher than when no permeable zone is present. A higher vertical flow outside the bentonite-sand barrier can be expected in all cases where a permeable zone may supply the rock around the bottom part of the bentonite-sand barrier with flowing water.

Should the fissured zone be one where substantial movements have been taking place, the feasibility of constructing a WP-Cave at that site can be questioned. Such major faults must be identified in the site investigation phase.

Compared with the cave, the KBS-3 geometry is mainly horizontal and can be adopted to existing zones irrespective of their orientation.

Conclusions

The large vertical extent of the WP-Cave, or its hydraulic cage, means that there is a high probability of the repository coming into direct contact with one or more conductive subhorizontal zones. This might lead to releases of radionuclides directly into the fastest migration channels reaching the WP-Cave. Consequently, the concept will not be able to take advantage of the full potential of the geosphere as an natural barrier against radionuclides reaching the biosphere. Should the existence of these zones not be acceptable from the construction or safety point of view, the availability of sites could be severely limited.

3.4 TECHNICAL FEASIBILITY

The WP-Cave was developed originally by a construction company and later by a major mining company in Sweden. It was considered by them to be feasible to construct, requiring only normal adaption of existing technologies to new circumstances. The disposal units of KBS-3 have been built on half scale.

There is one major difference between the WP-Cave and KBS-3 as far as feasibility of construction is concerned. The WP-Cave structure has never been built before. The most similar type of project is cut-and-fill mining in narrow stopes. Most probably some kind of pilot construction of a complete WP-Cave unit will be required in order to gain acceptance for the concept. This is not judged to be the case for the KBS-3 concept since it is based on the distribution of the waste among several thousand separate but identical units for waste isolation.

3.5 FACTORS AFFECTED BY ELEVATED TEMPERATURES

3.5.1 Geochemical Considerations

Rock Alteration

In the KBS-3 concept the canister surface would reach a maximum of 80°C and the rock 60°C. This limitation was applied to keep the bentonite buffer well below a temperature where any chemical instability could be expected /3-4/. The maximum temperature in the sand-bentonite buffer in the WP-Cave concept will reach approximately the same level as in KBS-3 for the same length of time. The temperature increase would cause an increased rate of mineral alteration in the granitic rock mass of the repository. This can be modelled by geochemical calculations and studied in laboratory autoclave experiments or by observation of hydrothermal mineral alterations in nature. The geochemical modeling consists in essence of equilibrium calculations, which do not to include rates due to reaction kinetics. The calculations primarily serve the purpose of indicating the tendencies of certain geochemical reactions to occur in the system and predicting the reaction products. The hydrothermal period of the repository is short compared to most geologic events and long in comparison to laboratory tests.

The alteration of granitic rock around a KBS-3 repository has been simulated by geochemical calculations /3-5/. The temperature in these simulations was set as high as 100°C in the near field rock. It was found that the tendency will be for porosity to decrease, mainly due to the production of secondary clays. These calculations were later repeated with the same initial conditions and temperatures up to 150°C /3-6/. The same tendency of porosity to decrease was found. This has also been reported for similar evaluations of heat storage concepts /3-7/. The general conclusion is that in closed systems or low flow systems such as the near field of the KBS-3 concept, the initial increase in porosity due to mineral dissolution will be relatively unimportant quantitatively. The decrease in porosity due to clay mineral formation and calcite precipitation will dominate. The calculations are based on observations of reaction paths in hydrothermal natural rock-groundwater systems.

An estimate of reaction rates was made for KBS-3 near field rock and 100°C /3-5/. The expected porosity decrease was roughly between 2 and 20% after 100 000 years. The temperature and duration of the thermal pulse are considerably shorter in the KBS-3 concept, so a noticeable effect should not be expected. For KBS-3, the temperature will reach a maximum after about 50 years and will have declined back to ambient after about 10 000 years. For the WP-Cave concept the duration of the thermal period is roughly the same as in KBS-3 but the temperature is well above 100°C. Therefore, the possibility of a noticeable change in porosity cannot be excluded for the WP-Cave concept.

The importance of this reduction of porosity is difficult to evaluate, since it depends on where it occurs. A porosity reduction over long times in the hydraulic cage might impair its function. If the porosity reduction is limited to inside the bentonite layer, the internal water circulation might be reduced, with a longer thermal period close to the canisters. At the same time the transport of radionuclides to the bentonite layer might be lower. There is, however, the risk that the porosity reduction might be dependent on the flow rate of the circulating water, and thereby only reduce matrix diffusion into solid blocks.

Water Chemistry

The results of autoclave experiments with granite and water have been reported for temperatures up to 150°C /3-8/. The study aimed at heat storage conditions in rock. The experiments confirmed the general view that no dramatic change in water composition should be expected due to thermal conditions. One exception is silica, whose solubility is highly temperature-dependent. Silica showed an increase in concentration from about 2 to 60 mg/l in a mixture of tap water and granite, when heated from ambient temperature to 150°C.

The increased solubility of silica will not change the overall porosity of the rock to any major extent. The long-range net effect will rather be a reduction in porosity due to the formation of secondary minerals with lower density, as has already been mentioned in the previous section. However, circulating water in combination with a heat gradient may cause a transport of silica outwards from the canisters, tending to increase the porosity in the sand and rock near the canisters and decrease the porosity in cooler regions towards the bentonite barrier. A mass transport in the opposite direction may be expected for calcite, which is a common fracture-filling mineral that becomes less soluble as temperature increases.

Radionuclide Chemistry

Equilibrium calculations have been performed on the temperature dependence of uranium and plutonium solubility under WP-Cave repository conditions /3-1/. For both elements it was found that solubility decreases with temperature. This is of special importance for uranium, which in the form of uranium dioxide constitutes the solid matrix of spent nuclear fuel, containing most of the activity. It was also concluded that in comparison to redox conditions, the influence of temperature on uranium solubility is negligible.

Refined calculations of uranium diòxide solubility have recently been performed and the results compared with a series of experimental measurements /3-9/. The temperature range studied was 25 to 300°C. From this study one can conclude that it is possible to make meaningful calculations of uranium solubility at 150°C. However, the uncertainty in the results is considerably greater than at 25°C. In comparison with uranium, the other actinides and technetium have less well established thermodynamic data bases. Therefore, present calculations of solubilities of these elements at 150°C are very uncertain.

Due to the relatively short-lived steel canisters suggested for the WP-Cave concept, a dissolution of radionuclides at high temperature conditions must be anticipated. This is in marked contrast to the KBS-3 concept, where the canisters will remain intact until normal background temperatures are restored. In the case of an initially damaged KBS-3 canister, the temperature will still not rise higher than 80°C.

Sorption, Diffusion and Colloid Formation

An increase in temperature generally increases the sorption coefficient /3-10/. In the long-term perspective, weathering reactions will generate secondary minerals such as clays, resulting in an increased sorption capacity. Further, mineral dissolution/precipitation, which is part of the reactions, will increase radionuclide fixation by co-precipitation. No adverse effect is therefore expected for radionuclide sorption in the WP-Cave repository due to the temperature increase. Diffusivity is not very temperature dependent.

Differences in temperature as well as differences in redox conditions could create mobile mineral particles containing radionuclides in the WP-Cave repository. However, the bentonite-sand buffer will be a shield against the escape of radio-colloids in much the same way as the bentonite buffer in the KBS-3 concept.

Natural Analogues

Conclusions regarding rock weathering, material stability, radionuclide solubility and mobility under hydrothermal conditions can be supported by observations in nature. Prominent examples of natural occurrences that can be used as thermal analogues are the Oklo uranium mine in Gabon /3-11/, the granitic rock body at Empire Creek Stock, Montana, USA /3-12, 3-13/, the hydrothermal alteration in the Auriat granite in the Massif Central of France /3-14/ and the Cigar Lake uranium deposit in northern Saskatchewan, Canada /3-15/. However, a detailed interpretation of thermal analogues also requires access to equilibrium constants and kinetic data for high temperatures.

3.5.2 Thermally Induced Internal Circulation

The thermally induced internal water circulation results in a faster transport of dissolved nuclides from the storage space to the inside boundary of the bentonite-sand barrier than would be the case if the water were stagnant and transport by diffusion were dominant. In this sense internal circulation is a mechanism that has a negative impact on radionuclide release from the repository.

Internal circulation will also ensure that the dissolved nuclides are well dispersed in the rock mass inside the bentonite-sand barrier. All fracture surfaces in contact with contaminated water are available for sorption and subsequent diffusion into the rock matrix. The effect of this for a non-sorbing nuclide will merely be a dilution in the water in the fractures and in the pores in the rock matrix. This water volume will be in the order of 10 000 to 20 000 m^3 for WPC 1100. Compared to the dilution obtained in the water volume in the bentonite-sand barrier, which is on the order of 105 000 m^3, the dilution inside the barrier is small.

For a low-sorbing element like tin with a sorption coefficient of 0.001 $m^3 \cdot kg^{-1}$, the whole rock volume between the storage space and the bentonite-sand barrier must be equilibrated in addition to the bentonite-sand barrier in order to reach a concentration that is ten times lower than the concentration obtained without considering any sorption in the rock mass. However, assuming a fracture spacing of 0.5 m, it will take approximately $8 \cdot 10^4$ years before this situation occurs. If the whole volume of rock and the bentonite-sand are in sorption equilibrium with a highly sorbing species like zirconium (sorption coefficient 4 $m^3 \cdot kg^{-1}$), the concentration will be about 30% of the concentration obtained without any sorption in the rock but accounting for sorption in the bentonite-sand barrier. The time required to equilibrate the whole rock volume is in this case about $3 \cdot 10^8$ years.

Over longer periods, the effect of internal circulation in terms of keeping fracture surfaces in contact with the contaminated water will have some impact on the release from the bentonite-sand barrier of sorbing non-solubility-limited nuclides. However, because of the fast transport from the storage space to the inside boundary of the bentonite-sand barrier, large amounts of the nuclides will escape from the repository before the effect of sorption on the inner rock mass becomes sufficient.

Another aspect of internal circulation in combination with the increased temperature in the repository is the alteration of the rock minerals that might occur. Two effects of the alteration can be identified. Firstly, alteration of the rock minerals will result in a conversion to secondary minerals such as clays, which normally have a higher sorption capacity than the primary rock minerals. Secondly, due the temperature dependence of solubilities, a transfer of minerals from one location to another might occur. Silica, which has a solubility that increases with increased temperature, will dissolve near the centre of the repository and be transported radially outwards with the circulating water and precipitate. Calcite has a solubility that decreases with increased temperature. A transfer of calcite in the opposite direction to the transfer of silica is then expected. The resulting effect of this redistribution of minerals in terms of opening and closure of fractures in different parts of the repository is therefore difficult to predict, especially since thermal expansion of the rock further complicates the situation.

In summary it can be concluded that more data regarding the internal rock mass, such as permeability, available sorption surfaces, sorption capacity etc, are needed before the effects of internal circulation can be fully quantified. The limited information available today indicates that the barrier effect of the internal rock mass will be small due to water circulation. This would not be the case if transport through the rock mass were dominated by diffusion. The difficulties in describing the complex situation in the internal parts of a WP-Cave repository must also be considered a negative aspect in comparison with a KBS-3 repository, where none of the above problems exist.

3.5.3 Thermally Induced Groundwater Flow

The calculated flow out through the top part of the bentonite-sand barrier is initially about ten times higher than the assumed natural flow in the bedrock at that depth, when hydraulic properties of rock, backfill in inner repository and bentonite-sand mixtures are assumed to be in accordance with the low-flow-through case. After 3 000 years the velocity will have decreased to about one fifth of the peak velocity.

The major factor affecting this flow rate is the hydraulic conductivity of the bentonite-sand barrier. The potential for improvement is a reduction by a factor of ten /3-1/, if no major adjustments are made in the design

The hydraulic conductivity of the inside is much higher than that of the virgin rock, and the inside thus represents a negligible resistance to flow compared to the bentonite-sand barrier.

The hydraulic cage has a negligible impact on the thermally induced flow through the cave.

A highly conductive zone in the bottom region of the cave has been found to increase the vertical flow velocity along the outer wall of the bentonite-sand barrier. No increase has been observed in the flow velocity through the barrier. The occurrence of such a zone is a matter of concern and is one important issue in the characterization of a potential site for a WP-Cave.

3.5.4 Conclusions

In most repository concepts, criteria for maximum temperatures in the repository are established as a compromise between two conflicting wishes. On the one hand there is a desire to concentrate the waste to get a small and inexpensive repository. On the other hand, a situation should be avoided where the temperatures are so high that chemical phenomena important for the safety of the repository cannot be evaluated with confidence. In the KBS-3 concept the limit was set at 80°C to allow a margin up to 100°C. It seems possible to raise that limit today. In the WP-Cave the limit was set at 150°C. Today it looks like this limit may cause substantial uncertainties in the solubility calculations. Both repository concepts can, with cost consequences, be adapted to other temperature criteria.

If the WP-Cave design is changed so that the canister material is copper, as in KBS-3, the service life of the canisters would most probably postpone the start of leaching until after the temperature pulse. This would substantially reduce the uncertainties due to solubility and due to the induced thermal gradients. A thorough investigation of the thermodynamic stability of copper in the WP-Cave environment has, however, not been carried out, cf 3.6.3.

The relatively high temperature at the steel canister surface will increase the corrosion rate. In contrast to copper used in the KBS-3 canisters, iron is not thermodynamically stable in water (See Subsection 3.6.3).

The temperature development in the bentonite-sand buffer of WP-Cave will be roughly the same as in KBS-3.

The increased solubility of silica will initially increase porosity in the rock inside the bentonite-sand shield. This initial porosity increase is insignificant. The increased weathering rate due to heat will reduce porosity in the long term perspective. The temperature gradient in the rock inside the bentonite-sand shield will cause a redistribution of material due to temperature-dependent differences in solubility (and advective-diffusive transport). Silica will tend to be transported outwards and calcite inwards to the canisters. The influence of temperature-induced weathering on water flow conditions inside the WP-Cave remains to be evaluated and quantified.

Radionuclide sorption on rock minerals generally increases with temperature. Weathering reactions due to heat are expected to increase sorption and promote fixation. However, in order to quantify this for use in the safety analysis as a mechanism for increasing the retention of released radionuclides, additional experiments are needed.

Radionuclide solubilities will be affected by the high temperature. In contrast to KBS-3 the WP-Cave canisters are not expected to out last the thermal pulse. The effect of 150°C on the solubility of actinides and technetium is expected to be small compared to e g redox conditions. However, this has not been fully evaluated yet. It is considered possible to make a reliable estimate of uranium solubility at 150°C and under WP-Cave conditions. It will be less accurate than the estimate at ambient temperature, but still useful. Thermodynamic data for other actinides and technetium at high temperatures are inadequate at present to make meaningful calculations of solubility and speciation at 150°C possible for the WP-Cave concept. Laboratory measurements to supplement these data would be too large an undertaking for SKB and the Swedish laboratories.

Natural analogue investigations can be used to support conclusions and modeling results of hydrothermal effects on canister material, bentonite, repository rock and radionuclide solubility and mobility. Conclusions regarding long term performance can be validated, provided that good enough geochemical data are available for high-temperature conditions.

Experience from the evaluations of radionuclide chemistry inside the WP-Cave as a function of temperature and time points to the necessity of an extensive program of laboratory measurements and natural analogue studies in order to evaluate radionuclide release and transport in this high-temperature situation with any degree of confidence. The amount of work involved is estimated to be too great to be carried out on a national basis. Laboratories in other countries would have to be involved.

3.6 CHARACTERISTICS OF THE BARRIER SYSTEM

3.6.1 General Principles

The final repository shall keep the release of radioactive substances to the environment within the dose limits set by society. This is achieved with barriers that isolate the waste from the biosphere and limit any release from the repository so that unacceptable concentrations will not occur in the biosphere. The assessment of the safety of the repository is based on the performance of the barriers under relevant environmental conditions. Barrier performance is evaluated by means of models of processes involved in mass transport, based on solubility limitations and the long-term stability of the barriers.

Acceptance of the assessment results is dependent on the conceptual understanding of the processes affecting barrier performance, the availability of good and tested calculation models and the quality of the available data. The uncertainty in the analysis as regards the data and the calculation models can in some instances be quantified.

A minimum requirement on the data on which the decision as to whether the repository is safe enough is based is that the decision should not be altered by expected uncertainties in the input parameters. Quantifiable uncertainties can be compensated for by using safety margins. To reduce the importance of uncertainties that cannot be quantified, the following principles can be formulated:

- The selection of a repository concept and the selection of barriers should be made bearing in mind the need for quantified safety assessments,
- Processes relevant to the safety of a barrier should be based on scientifically established data and knowledge,
- The long-term safety of the total system should be based on many independent barriers. In this case the safety of the repository will not be completely lost if the performance of one barrier is poorer than predicted,
- The repository should be arranged in such a way as to utilize the potential of the host rock as a safety barrier.

In view of the long time spans that are often involved in the acceptance criteria, it is advantageous if materials and other characteristics of the repository can be selected in such a way that information obtained from nature or natural analogues can be used.

The principles outlined above are intended to aid in quantifying the safety level of a repository. These principles can also be used to assess differences between the various alternatives investigated with regard to their potential for quantification. It must be kept in mind, however, that they are not the only criteria that can be applied when comparing alternative repository concepts (see e g Section 1.2).

In the following the safety barriers of the WP-Cave concept are discussed first with regard to their combined effect and then one by one.

3.6.2 WP-Cave Barrier System

The WP-Cave configuration incorporates six relatively independent barriers:

- the canister isolating the waste from the groundwater,
- the fuel matrix, with a low solubility,
- precipitation at the redox front,
- sorption on the inner surfaces of the repository,
- delay and sorption in the bentonite layer,
- diffusive uptake of nuclides into the water flowing past the buffer layer,
- delay and sorption in the geosphere.

As long as nuclide transport out from the repository is mainly by diffusion, performance assessments indicate that repository safety is dominated by four mechanisms:

- Slow oxidation by radiolysis and the low solubility of the uranium oxide matrix and of several other radionuclides limits the radionuclide concentrations that can occur in the water,
- Retardation due to sorption in the bentonite allows several short-lived radionuclides to decay,
- Low diffusivity in the bentonite barrier limits the rate of diffusion through the barrier,
- Slow diffusion into the water flowing past the outside of the bentonite greatly limits the amount of water that is contaminated.

If nuclide transport is dominated by flow through the bentonite, the safety of the repository is mainly guaranteed by the first two phenomena.

The expected service life of the steel canister, a few hundred years, is short compared to the diffusion time through the bentonite-sand barrier, and the sorption of radionuclides on the inner surfaces is relatively small compared to the sorbing capacity of the bentonite-sand mass. Furthermore, when a hydraulic cage is assumed around the repository, the radionuclides released from the cave will mainly be transported to the biosphere along the main transport route contacting the hydraulic cage (eg. a fissure zone with high permeability).

The performance assessments indicate that doses from radionuclides with high solubility and low sorption might reach acceptability limits and in some cases exceed them when released to a well recipient with poor dilution capacity. This is especially true in the flow-through situation at high thermal gradients.

The barriers are quite similar in the KBS-3 system:

- the canister isolating the waste from the groundwater,
- the fuel matrix, with a low solubility,
- delay and sorption in the bentonite layer,
- diffusive uptake of nuclides into the water flowing past the buffer layer,
- precipitation at the redox front,
- delay and sorption in the geosphere.

Here, however, the life of the copper canister and the dissolution time of the fuel matrix are of about equal magnitude, about one million years. The bentonite layer mainly has the role of reducing the release rates, not really delaying the travel time of the radionuclides to the biosphere. Although the redox front may be at different distances from the fuel in the two concepts, it has in principle the same role, to reduce solubilities.

As far as geosphere transport is concerned, the main difference is that the locations of the deposition tunnels and the actual deposition holes in KBS-3 can be selected with regard to the hydraulic qualities of rock, thereby avoiding the vicinity of the most unfavorable transport routes to the biosphere. Although the existence of a strong channeling effect cannot be excluded, this option to use only what is believed to be better parts of the rock enables the KBS-3 concept to better utilize the geosphere barrier.

Conclusions

Due to the high temperature, the fixed geometry of the design, and, above all, the heavy reliance on the bentonite-sand barrier, the WP-Cave concept offers poorer multi-barrier protection and entails greater problems in quantifying barrier performance.

However, there is nothing in the system to prohibit the use of copper as a canister material in the WP-Cave as well. The more complicated chemical situation and the high temperatures in the cave might make it harder to guarantee an extremely long service life, but service life can be extended by several powers of ten compared to the steel canister. A longer canister service life would reduce radiolysis on canister penetration and temperatures during matrix dissolution and postpone the time of canister penetration to a point beyond when the flow has passed its peak. Such a change must of course also be taken into account in the cost calculations.

In the same way, a thicker bentonite layer could be employed in the KBS-3 system, at a higher cost and at the price of higher surface temperatures or less fuel in the canisters.

3.6.3 Canister Materials

Carbon steel has been chosen as the canister material for WP-Cave. This choice of material has many advantages in the fabrication and encapsulation phases of repository operation. However, in contrast to copper, which is thermodynamically stable in water, steel will corrode under oxidizing as well as reducing conditions. Under reducing conditions, corrosion will proceed with hydrogen evolution, cf 3.7.

Because of the differences in corrosion behavior and in environment, the expected service life of a steel canister in WP-Cave will be considerably shorter than that of a copper canister in a KBS-3 repository. There are many uncertainties concerning the rate of the anaerobic corrosion of steel in WP-Cave, but canister service lives of a few hundred years seem justified. The copper canisters in KBS-3 have an expected service lives exceeding 100,000 years. Due to this earlier canister failure in WP-Cave, more short-lived nuclides will also have to be considered in the safety analysis.

Naturally, copper can also be used as a canister material in WP-Cave. However, conditions in the cave have at present not been sufficiently analyzed to permit an estimate of the service life of a copper canister. For one thing the high temperatures, 150°C, together with the unknown but probably very high sulphur content of the water inside the cave, will make it more difficult to assess the extent of sulphate/sulphide corrosion of copper.

In the KBS-3 concept, the barriers are not completely redundant but are to some extent coupled. This interrelationship between the barriers is much more pronounced in WP-Cave. Nevertheless, it seems reasonable to assume that in WP-Cave as well, a copper container will have a considerably longer service life than a steel container, and will probably constitute the dominant barrier against radionuclide release from the repository for a very long time. This would reduce the requirement on the performance of the bentonite-sand barrier, which in the present design is by far the dominant barrier. Instead, the performance of the cave would be strongly dependent on the performance of the canister.

3.6.4 Bentonite-sand Barrier

The bentonite-sand barrier has several functions. It reduces the flow rate of water to the canisters, acts as a diffusion barrier and has good sorption properties for many nuclides.

In the KBS-3 design a 0.38 m thick compacted bentonite layer is used, whereas the WP-Cave is surrounded by a 5 m thick sand/bentonite mixture with only 10 to 20% bentonite. The density of the bentonite in the mixture is much lower. The amount of bentonite a nuclide will have to migrate through during its escape is, however, not very different. The apparent diffusivities in the compacted bentonite and in the sand/bentonite do not differ very much for several of the major nuclides, and because the thickness of the layer is greater in the WP-cave design it might be expected that the thicker barrier would stop some more nuclides from escaping.

In the KBS-3 design, it has been shown that the flow rate of water through the backfill is so small under normal repository conditions that the transport of dissolved nuclides out of the repository is dominated by molecular diffusion. In the WP-Cave barrier, the hydraulic conductivity is higher and the hydraulic gradient over the bentonite is considerably higher. The latter effect is caused by the fact that the rock inside the WP-Cave is highly conductive and in practice presents very little resistance to the flow. The transport of nuclides through the barrier by advection will be comparable or higher than by diffusion. Even during the early period when the temperature is high in the WP-Cave, there is a considerably increased flow through the interior.

For the early non-steady-state period the following may be noted. The service life of the canister is much shorter in WP-Cave and some of the nuclides with shorter half-lives, notably C-14, Cs-137, Sr-90, Am-241, Am-242, Am-243, Pu-240, will not have decayed before they start escaping. It was found that, out of these, only C-14 and Pu-240 do not decay to insignificant levels during their diffusion in the backfill. Several other nuclides also decay considerably during their transport in the backfill /3-1/. The initial canister failure scenario in KBS-3 indicated that the time delay caused by the bentonite layer during the non-steady-state phase is smaller than the delay caused by the bentonite layer in WP-Cave.

When the non-steady-state phase has passed in the KBS-3 design and the concentration profiles in the bentonite have become steady-state, the barrier limits nuclide transport to the moving groundwater in two ways. The hydraulic conductivity is so small that transport occurs only by diffusion, and the flow rate of groundwater around the bentonite layer is so slow that concentrations build up, reducing the concentration gradient driving the diffusion. The latter phenomenon, diffusive resistance into the moving groundwater, dominates, and the diffusion resistance in the backfill plays a negligible role under the conditions evaluated in KBS-3. The water flow rate in the rock outside must increase by several orders of magnitude in order for the diffusion resistance in the clay barrier to make any difference.

In the high-flow-through case, release from the WP-Cave is dominated by the flow through the repository. The release of nuclides will then be proportional to the flow rate through the cave. In summary, the main role of the bentonite barrier in the WP-Cave is to limit the flow of groundwater through the cave.

Due to the high conductivity in the center of the cave, the conductivity of the bentonite-sand layer will determine the flow through the cave. Even a 5 m thick barrier will permit so high a flow rate that the release of radionuclides will be dominated by advection.

If the hydraulic cage were to reduce hydraulic gradients in the cave area substantially, the flow through the cave after the thermal period would be substantially reduced.

Under normal circumstances the bentonite will not be eroded by passing water. If the concentration of dissolved salts should for some reason decrease considerably, the bentonite will not form a stable gel anymore and may be carried of by the flowing water. The higher flowrates, especially locally in the cave, compared to KBS-3 may cause this to be a more important factor.

3.6.5 The Hydraulic Cage

The aim of the hydraulic cage is to radically decrease the hydraulic gradient over the cave, thus reducing the groundwater flow to a minimum. The concept is identical to that of Faraday's cage.

The hydraulic cage consists of a system of interconnected boreholes located 50 m outside the bentonite-sand-barrier. The cage is to be drilled and drained prior to the excavation of the cave. The intention is to drain the rock mass inside, creating dry conditions during the excavation of the slot for the bentonite-sand barrier. Any water transport through the hydraulic cage can be detected during excavation and backfilling of the slot. If hydraulic pathways are detected additional boreholes can be drilled.

The basic calculations showing how the hydraulic cage works were carried out under the assumption of a homogeneous hydraulic continuum /3-1/. Their results indicated that the cage effectively reduces the gradient through the repository volume even if some of the boreholes are plugged for some reason. The calculations also showed that the cage effectively collects the regional groundwater flow for an area considerably larger than the cross-section of the cage, resulting in high flow rates in the borehole system.

The present work has pointed at three problem complexes associated with the hydraulic cage:

- The fracture system is poorly approximated by a homogeneous continuum. The effects of channeling and other inhomogeneities must be considered,
- The thermally induced gradient caused by the heat from the decay of the spent fuel must be considered,
- Possible clogging of the borehole system in the cage over the long term could be important.

The effects of the inhomogeneities in the fracture system have been investigated /3-1/. The study shows that the hydraulic cage may play an important role if the transport distance between the cave and the biosphere is small. For large distances the effect of the hydraulic cage is less important.

The calculated release from the repository is reduced by a factor of 3—10. However, the influence of the cage on the regional flow means that more fast channels are involved in the far field flow. The authors of /3-1/ conclude finally that limited knowledge of the characteristics and frequency of the channels makes a definite and generic determination of the effect of a hydraulic cage impossible.

The thermally induced flows have been studied /3-1/. The results show that the hydraulic cage has virtually no effect in reducing the thermally induced flows through and around the cave. The vertical flow through the bentonite-sand barrier reaches a maximum after about 300 years. After 3000 years the flow is reduced by one order of magnitude.

The length of the period of thermally induced flow is, however, limited, and if other barrier systems are effective during this period the failure of the hydraulic cage to prevent thermally induced flow may be of less importance.

The third question, the possible long-term plugging of the cage boreholes, still remains to be answered. The flow velocities in the boreholes are high both with respect to both the passage of the natural flow around the cave and the thermally induced flow. It is not clear whether the boreholes will eventually be plugged, causing the effect of the hydraulic cage to be lost. From a material balance point of view this is quite possible.

3.6.6 Transport and Dilution of Contaminants in the Geosphere

Introduction and Background

During their transport through the bedrock and upon their release into the biosphere, the radionuclides are diluted. The degree of dilution is highly dependent on site- and system-specific factors, such as:

- The construction and layout of the repository.
- The hydraulic characteristics of the bedrock.
- The type and characteristics of the groundwater recipient in the biosphere.

The repository will be located in a region where the natural flow is essentially downward. The natural transport pathways will eventually turn upward and the contaminated water will finally reach a creek, lake or a well. The travel distance can vary and it is conceivable that under some circumstances the flow paths will be rather short, on the order of a distance equal to the depth of the repository.

In this subsection a special case will be discussed where a very small dilution could occur. It can occur when all the radionuclide-contaminated water from the repository flows to one single well with a low water production rate.

Release from the Repository

Two important release modes have been identified. The first is by water flowing through the repository. The flowing water will carry the nuclides out of a concentration equal to that inside the clay barrier at that time. The "flow-through" release mode is clearly dominant during the thermal period, and the flow rate through the repository could be as high as 70 m^3/year in the high flow-through case. After the thermal period the release is also dominated by the flow-through mode and the flow rate of contaminated water is about 11 m^3/year.

The second release mode is by diffusion through the bentonite/sand and into the slowly moving water passing by the outside. The release is limited by the rate of diffusion into the water moving in the narrow fractures in the rock. The release rate in this case can be visualized in the same way as in the "flow-through" case. The equivalent water flow rate contaminated to the concentration which is present in the repository any particular year could be expected to be about 0.9 m^3/year. In the high-flow-through case this release mode is negligible compared to flow-through release. The clay does not actually act as a diffusion barrier. The above figures apply to the case where there is no hydraulic cage present. The cage will reduce the release. This is further discussed below.

In KBS-3, the "diffusion mode" is the dominant release mechanism.

One of the crucial questions is whether the contaminated water is mixed with and diluted in larger volumes. For flow in a homogeneous porous medium with small pores, transverse dispersion and dilution could be readily calculated. Rock is not such a medium and flow takes place in fractures where channeling occurs. The channels are few and far between, and over short distances at least the channels may not intersect and mix their waters, although some dilution may take place by diffusion into stagnant or slowly moving zones of water.

Influence of the Hydraulic Cage

If the contaminated water enters the water flowing in the hydraulic cage, the diffusion distances are shorter and the residence times longer. The presence of the cage will reduce the flow rate of contaminated water and give an overall smaller release, cf 3.6.5. The smaller flow rate will, however, be diluted in the water flowing through the cage.

The flow rate of water in the cage could be estimated to be on the order of 27 m^3/year, allowing for the increased flow of water to a considerably more permeable region in a rock. The cage ensures that the contaminated water will flow in the channels with the highest flow rates, which are probably also those that will have the shortest residence times in the far field.

Well Hydrology

The KBS repository is assumed to be located at a depth of about 500 m, consisting of a horizontal network of tunnels covering about 1 km^2 and with deposition holes down to about 10 m below the tunnel floor. With a specific discharge of $1.3 \cdot 10^{-5}$ $m^3/m^2 \cdot$ year, the total amount of water passing the repository would be 10—30 m^3/year, or of the same magnitude as the contaminated flow from the WP-Cave. Only a portion of the water that flows through the KBS-3 repository is contaminated, but if the same flow pattern as above is assumed, all the small flow tubes from repository can be contained in a larger flow tube and can end up in a single well in the same fashion as for the WP-Cave case.

There are, however, a number of important differences between the two repository concepts. In the KBS-3 repository, there are nearly 6000 canisters in individual holes. The channels in the rock run a small risk of intersecting a single canister. This possibility was not considered in the analysis of KBS-3 because knowledge of channeling did not exist at that time. A simple geometrical calculation assuming that every channel has a catchment width of 1 m shows that there is only one chance in about 50 that a channel will intersect a canister hole, whereas the WP-Cave, because of its large extent, is bound to intersect all channels in that rock volume. There is also a difference in the minimum distance between the bottom of the well and the top of the repository, about 100 m for the WP-Cave and about 400 m for KBS-3. The longer water residence times in the lower rock portions will allow more transverse diffusion and may let some more nuclides move into water paths that may not be in the stream tube leading to the well. It is at present not possible to quantify these differences for a general case, but it might be possible to make more definite predictions for a specific site.

A further fundamental difference is that the tunnels in KBS-3, unlike the hydraulic cage in WP-Cave, are not made for the purpose of establishing hydraulic contact between the water- bearing fractures or channels. In the backfilling and sealing stage, measures will be taken to reduce the hydraulic interconnections created by the excavation work, thereby further reducing the risk that one well will affect the whole repository. If the seven WP-Caves needed for the total amount of Swedish spent fuel are not sited close to each other, the distribution of waste in different localities will have a similar effect.

Discussion and Conclusions

There is no obvious difference between the probabilities of a well drawing in the contaminated water in the two types of repositories. It is not difficult to imagine conceivable flow situations where all or most of the contaminated water from a repository is drawn into a well and the only dilution that takes place is in the water pumped from the well.

In the KBS-3 repository with its many separate canisters, the channels run a small risk of intersecting a canister.

3.7 GAS EVOLUTION

The migration of hydrogen gas through the bentonite-sand barrier and the bedrock has been analyzed /3-1/. The sand-bentonite barrier is impervious enough to act as a gas-tight shield, like the "cap rock" in natural gas reservoirs. A material-specific "critical" pressure must be reached in the gas cushion before the gas can penetrate the bentonite-sand barrier. This cushion is assumed to be formed in the top part of WP-Cave. It grows in volume until breakthrough occurs. During growth of the gas cushion, water is displaced into the bentonite-sand barrier in volumes equivalent to the gas production. Once gas has escaped through the barrier, water starts to flow back into the repository.

The critical pressure for a bentonite-sand mixture with 50% bentonite (by weight) is considered to be between 0.5 to 1.5 MPa above the existing hydraulic head. This means that a theoretical gas pillar of 50 to 150 meters may develop. For a 10% mixture the critical pressures are lower, about 0.05 to 0.1 MPa overpressure.

The migration through rock is dependent on the existence of fissure systems and connecting transportation paths as well as width and other properties of the fissures. For rock masses with hydraulic conductivities common at depth of 300–500 m in the

Swedish bedrock and with fine (capillary) fissures only, evenly distributed, the gas transportation capacity could be too small for evacuating the gas in a WP-Cave. If so it is considered very likely that the gas pressure increases, water is expelled and the water table lowered to a level where steel submerged in water produces no more hydrogen gas than can be transported away. The effect from any existence of "channels" in the rock has not been analyzed so far.

A very high gas cushion may be acceptable as long as the gas does not reach the top canisters, in which case the consequences are difficult to assess. A 50% bentonite mixture might exceed this limit, but if a lower-grade mixture is used in the cylindrical part of the bentonite-sand barrier, the gas should not be able to expel water below the top canister level. The bedrock must have a sufficiently high gas migration capacity.

The range of lowest-highest water level in a build-up — breakthrough sequence in the clay barrier could be very narrow if pores in the bentonite-sand mixture do not close completely once they have been opened. A steady-state water table level could be developed, which would have the advantage of maintaining the same water volume inside the bentonite-sand barrier.

The rate of hydrogen production during the anaerobic period is uncertain. Conservatively, very high corrosion rates, in the range of 150 μm/year, may have to be considered. In addition to the canisters, WP-Cave will also contain a substantial amount of construction steel. With this higher corrosion rate, the hydrogen production rate has been estimated to be 74 000 m^3/year. However, this may well be a considerable overestimation, the actual hydrogen production rate being one to several orders of magnitude lower. This large uncertainty together with the question of gas evacuation through the rock makes the evaluation of the WP-Cave very difficult with present knowledge.

3.8 IMPACT OF MOVEMENTS IN THE EARTH'S CRUST AND HUMAN INTRUSION

3.8.1 Movements in the Earth's Crust

In the long-term perspective it is useful to understand the behavior of a repository in the event of possible movements in the earth's crust. Three earthquakes have occurred in Sweden during the last five years with a magnitude larger than 4 on the Richter scale. Their epicenters have, however, been deeply located, about 15—20 km below the earth's surface.

The effect of an earthquake has not been analyzed for the WP-Cave structure. It can, however, be noted that an acceleration of 0.1 g, which is the design criterion for the last two nuclear power stations built in Sweden, would not cause any damage to a WP-Cave structure.

In the case of WP-Cave the bentonite-sand barrier acts as a shock absorber and cushions the effect on the interior. With a totally enveloping bentonite-sand barrier, no constricted points are present where an impulse could lead to a permanent crack. Even though water has a good shock absorbing effect, canisters and fuel will probably be less affected if the interior openings are backfilled with crushed rock or finely ground rock.

Serious damage may occur if an earthquake results in major movements in the bedrock along fissure planes. From a theoretical point of view, a shear movement of a couple of meters along a plane that intersects the cave should not completely disable the isolating function of the barrier. The function of the hydraulic cage would probably be affected to only a minor extent.

The KBS-3 concept differs from the WP-Cave concept in that it is possible to avoid placing canisters in or near zones where rock movements have a higher probability of occurring. If a movement occurs, the scattered canister placement means only a limited number of canisters will be physically affected. The WP-Cave repository might have to accept that zones of potential movement are in contact with two of the repository barriers, the hydraulic cage and the bentonite layer. However, the thickness of the bentonite layer 5 m protects the inner parts of the repository against larger movements than the KBS-3 concept can withstand.

3.8.2 Human Intrusion

A basic distinction must be made between intentional and unintentional intrusion. Intentional intrusion may have the purpose of repairing or changing the repository system or retrieving and utilizing materials deposited. Future man, like man today, must have full freedom and the full responsibility for any deliberate action taken.

Unintentional intrusion assumes that knowledge of the site and the content of the repository has been lost. This is considered to be highly improbable for any kind of repository if there is a will to preserve the records and the information is available in many internationally distributed centers.

For a direct intrusion to constitute a hazard to man it must be assumed that the nature of the waste is not observed. Based on the unusual characteristics of any repository construction, and the relative ease of detecting radiation, such a situation is considered to be of such low probability that any difference between the two investigated concepts would be insignificant.

Due to the shallower depth of the top of the repository in the WP-Cave concept and the existence of the hydraulic cage, the probability of a well utilizing the water near the repository might be somewhat higher. However, the consequences of this have already been taken into account in the low dilution well scenario.

3.9 UNCERTAINTIES AND CONFIDENCE ISSUES

3.9.1 General Considerations

Assessing a disposal concept is closely related to forecasting the repository performance in time. Large inherent uncertainties are involved in the performance assessment since very large time spans and complex processes are involved. Repositories are also "one-of-a-kind". Their future behavior must be predicted by analyzing sub-processes and by coupling and extrapolation of those processes.

The confidence that can be placed in the assessment is largely dependent on the predictability of the system analyzed, how well it lends itself to generalizations such as modeling etc.

Different categories of uncertainties can be identified, more or less overlapping:

- uncertainties associated with external factors, i e site evolution in time regardless of whether the repository is there or not (geodynamics, glaciation, the biosphere etc),
- uncertainties associated with internal factors, i e repository performance given the external circumstances (e g radionuclide migration),
- uncertainties in the quantifications made (input data to models),
- uncertainties associated with more or less concealed conceptualizations and assumptions in the analysis,

– generic uncertainties (independent of repository design and detailed siting),
– uncertainties specific to the repository concept, etc.

An attempt will be made below to take these different categories of uncertainties into account, with an emphasis on those judged to be most important. The principal distinction is made between generic and concept-specific uncertainties. The concept-specific ones are of crucial importance in the assessment of the WP-Cave relative to other concepts. Generic uncertainties are discussed first, however, to provide a more complete picture and to find out whether these uncertainties are in fact greater than the concept-specific ones.

3.9.2 Generic Uncertainties

Future Geological and Biological Evolution

The future geological and biological evolution of a disposal site is definitely a key factor in assessing of the possibilities of describing repository performance in time. Factors that have to be taken into consideration include

- recurrent glaciations at the site, as a result of which
  - the geohydrological situation can be changed drastically, particularly the gradients on a local and sub-regional scale. Very small movements in the bedrock can also be expected to alter the properties of the fractures such as apertures, fracture fillings etc,
  - the biosphere is recurrently altered;
- very-long-term tectonic movements (e g uplift and subsidence) and erosion processes, as a result of which
  - the integrity of the repository can be threatened. The very long term is not longer than the time periods actually considered in the KBS-3 radiological modeling;
- the biological evolution of species, including homo sapiens, as a result of which
  - the extrapolation of the exposure pathways analysis and dose calculations to extended periods of time is futile.

These uncertainties are not related to the disposal concept as such; they are as important for the KBS-3 concept as for the WP-Cave.

Groundwater Transport Modeling

Even under the assumption of static external conditions, there is great uncertainty involved in the modeling of groundwater movement and radionuclide migration in fractured rock types such as gneiss and granite.

This has to do with the fact that, in granitic rock, only a small portion of the rock volume is involved in establishing the groundwater flow pattern, with very large less-pervious blocks between the flow paths. The three-dimensional flow patterns are mainly studied by observations in boreholes, which are, essentially, topologically one-dimensional. As a consequence, interpreting these observations is not so simple.

There is presently much debate as to the applicability of various conceptualizations in the modeling of groundwater flow and radionuclide migration in fractured media, one extreme being the representative elementary volume / continuous medium approach, the other being the more conservative channeling approach used in the WP-Cave performance assessment. Different conceptualizations give very different computational results, and a large, unquantifiable conceptual modeling uncertainty must be accepted at the moment. It is not related to the disposal concept as such, it is specific to the geologic medium.

Radiolysis and the Leaching Process

Another generic uncertainty issue is associated with the effects of radiolysis of water in contact with the fuel pellets, particularly from alpha radiation. Important factors are the redox conditions in the near field of each canister after the radionuclides have started leaking out and the oxidation of the fuel. The products of oxidation have a different crystalline structure than UO_2. The change in structure may cause a release of soluble radioelements trapped in the UO_2 matrix.

The uncertainties involved in trying to model the radiolysis-oxidation-leaching process and the redox conditions in the near field could have different implications for different disposal concepts. But they are likely to be of great importance, in any disposal concept.

Release into Slowly Moving Water by Diffusion

Performance assessment of the near field typically includes modeling of the mass transfer rate to the near-stagnant or slowly moving groundwater outside the engineered diffusion barrier of the repository. This type of modeling requires the same kind of conceptualization as the geohydrological modeling described above. The near field model used for WP-Cave as well as for KBS-3 assumes either flow in a continuous, porous medium or flow in individual fractures.

The difference in release rates between the porous flow and fracture flow concepts is not great. In WP-Cave there will be flow through the repository and nuclides will be released by the flow mode as well as the diffusion mode. For the KBS-3 repository only the diffusion mode will be active.

Well Hydrology

An assessment of the radiological consequences of radionuclides entering the biosphere is highly dependent on assumptions regarding the so called primary recipient. Particular uncertainty is associated with wells as primary recipients. Issues which must be resolved in the assessment include assumptions as to the extraction rate of water from the well, the fraction of the plume of contaminants actually headed for the well etc.

It would seem possible to construct scenarios with one hundred percent of the activity traveling to a well with a small extraction rate, leading to a relatively high concentration of radionuclides in the water. It is difficult to determine which assumptions are reasonable and which are not. The associated uncertainty is considerable, and very little dependent on the disposal concept.

3.9.3 Concept-specific Uncertainties

The uncertainties specific to the WP-Cave concept have been discussed in previous chapters and sections. An attempt is made here to examine those uncertainties more closely and compare them with the uncertainties associated with the KBS-3 concept, whenever possible.

Temperature

The relatively high temperature in WP-Cave, as compared to KBS-3, introduces uncertainties in the performance assessment of the former. This has to do with the fact that little geochemical and thermodynamic data is available for those temperatures. There is little reason to believe, however, that very dramatic differences occur, although the possibility can not be ruled out entirely.

Canister Life

The actual corrosion rate pertaining to the WP-Cave canisters is difficult to determine, mainly due to the fact that it is controlled by the chemical species in the near field of each canister. As pointed out above, there are uncertainties as the nature of the chemical environment.

On the other hand, the steel canister is short-lived in any case and a few hundred years more or less are not so important. The durability of the copper canister in the KBS-3 concept is also dependent on the near field chemical environment. In this case the actual life of the canister is more important, and is a crucial part of the concept. Uncertainties (if any) seem, in a way, to be more important for the KBS-3 concept, but on a higher base level of safety than for WP-Cave.

Gas Evolution

A problem specific to the WP-Cave concept is the evolution of gaseous hydrogen from corrosion of steel in the construction, including the canisters. As has been discussed in previous chapters, the amounts could be considerable. The gas has to find low pressure outlets through the buffer and the surrounding bedrock so that the structure of the WP-Cave will not to deteriorate due to the buildup of high pressures. The backfill and sand-bentonite barrier must be designed to provide these outlets, hence the relatively high hydraulic conductivity of the clay barrier in the WPC 1100.

Too low a gas permeability in the barriers could cause repeated high pressure buildup and subsequent relief through the sand-bentonite, which in turn could possibly create shortcuts to groundwater flow and diffusion of radionuclides. This must be considered an uncertainty with the WP-Cave at present.

Backfill Integrity and Sorption Properties

The WP-Cave near field modeling relies to some extent on the assumed favorable sorption and diffusion properties of the material used as backfill for the tunnels containing the canisters. The combined effect of diffusion and sorption cannot be assumed unless the mechanical integrity of the backfill is guaranteed.

Doubts could be raised as to this integrity. Movements in the walls of the tunnels due to excavation could jeopardize the favorable properties of the backfill. This kind of uncertainty cannot be ruled out entirely. There is no analogue in the KBS-3 concept.

Bentonite-sand Barrier Long-term Integrity

The non-distributed WP-Cave design, where the spent fuel is emplaced in a small number of caves, two to seven, imposes strict requirements on the quality of the engineered diffusion barrier, since large amounts of radioactivity must be contained within its boundaries. Hence, long-term integrity, as well as uncertainties with respect to this integrity, will be more important for the WP-Cave concept than for the KBS-3 design.

The Hydraulic Cage

The degree of confidence associated with the modeling of the hydraulic cage is related to some extent to the generic problem of modeling groundwater flow and radionuclide transport in the far field. Describing the rock as a continuum implies very favorable properties of the cage. Attempts to take discrete water-bearing structures into account, and to consider the long-term integrity of the cage, tend to cloud the picture. This is a type of uncertainty not encountered in the KBS-3 case. It is difficult to assess its importance.

3.9.4 Conclusions

The uncertainties specific to the assessed performance of the WP-Cave concept seem somewhat greater than those associated with the KBS-3 concept. The design is somewhat more complex, and there are some particular uncertainties having to do with the integrity of the engineered diffusion barrier, and, possibly, the generation of gas from steel corrosion, the hydraulic cage and high temperature. They seem to be overshadowed by the generic uncertainties, however, which seem more serious.

The concept-specific uncertainties are, however, not so great as to prevent the conclusion from being drawn that the differences in performance between the WP-Cave and the KBS-3 repository clearly favor the KBS-3 concept.

4 CONCLUSIONS

4.1 BACKGROUND

SKB is required to perform the research needed to evaluate various disposal methods and to select a site and repository system for final storage of spent nuclear fuel in Sweden. The KBS-3 concept, published in 1983, was reviewed in great detail by the both domestic and foreign groups of experts. Based on this scrutiny it was found by Swedish Government in 1984 that KBS-3 constituted a method for final disposal that was acceptable with regard to safety and radiological protection. Consequently, it has been selected as the reference concept for comparing various alternative methods for final disposal.

Most of the repository sites and barrier designs available for spent fuel disposal in Swedish bedrock are quite similar and require for their early development the same sort of basic information. As the designs develop there will be a need for greater detail in the evaluations. Priority must be given to those options that seem to have the best potential for provable safety and high cost effectiveness.

SKBs evaluation of the WP-Cave concept and comparison with the reference concept (KBS-3) is a part of this process of establishing priorities. It is basically an evaluation of the relative merits of a repository option based on concentrating the waste canisters in a small rock volume with an option based on distributing the canisters over a larger area.

As indicated in Chapter 1, a number of aspects are of importance in the comparison:

- Feasibility with regard to present technology.
- Radiological safety and potential for development.
- Cost effectiveness.
- Uncertainties and confidence.

The conclusions of SFG (the Integrated Performance Group) will be presented in the above order.

The WP-Cave concept was originally developed to allow a concentrated emplacement of spent fuel in an engineered barrier system. The rock volumes needed are small and the safety of the repository would be relatively independent of the quality of the rock. This in turn allows a cave to be sited e.g. close to each reactor site in Sweden.

The original thought was that a thick enough bentonite layer would limit the groundwater flow into the cave to such a degree that it would take hundreds of thousands of years for the cave to be filled with water. Another idea was that an early encapsulation and placement of the canisters in the cave would eliminate the need for an intermediate storage facility in the Swedish system.

As new data became available on the conductivity of bentonite, the safety potential of that barrier was reduced, and as the CLAB facility became operational the economics of the system changed.

The comparison made in this report is based on the Swedish situation today and on a repository concept that has subsequently been modified, especially in the course of the study performed by SKN.

In a number of situations during the evaluation, it was found that a specific phenomenon with a clearly negative influence could be avoided by changing the design or by substituting one material for another without changing the major features of the system.

For example, it was decided at an early stage to limit the maximum temperature to 150°C, to permit an evaluation of the rates of corrosion, fuel dissolution and redistribution of silicates and fissure-filling materials in the inner zone. Another change made was to fill the open spaces in the cave with a backfill before sealing the repository to limit the release rates of radionuclides from the canisters.

In most other cases the concept was evaluated as it was originally presented, although some suggestions have been made for changes. In this evaluation it has been considered more important to evaluate the principal differences between WP-Cave and KBS-3 rather than two specific designs. Performance has been analyzed to a depth needed to establish priorities.

4.2 TECHNICAL FEASIBILITY

The WP-Cave concept was developed originally by a construction company and later by a major mining company in Sweden. It was considered by them to be feasible to construct, requiring only normal adaptation of existing technologies to new circumstances.

The group accepts this evaluation and notes that site investigations will enable the feasibility of siting the repository at a particulate place to be determined.

4.3 SAFETY AND POTENTIAL FOR DEVELOPMENT

General

The radiological safety of the WP-Cave concept has been analyzed by evaluating the performance of the repository components and their interaction in a high-flow-through case.

This scenario consists of a series of unfavorable assumptions concerning drilled wells, transport channels from the hydraulic cage, groundwater flow through the repository and radionuclide solubility. The assumptions reflect the level of performance that is considered proven at this stage of analysis.

The results show that there are radionuclides in the repository that could be released to the biosphere in such concentrations that established radiation protection criteria (dose < 0.1 mSv/a) would be violated. With a refinement of the WP-Cave design a reduction of the peak doses might be achieved.

The group concludes that further development of both the concept and the models for evaluating its performance would be required to show that an acceptable safety level can be achieved by the WP-Cave concept.

Since WP-Cave, beside its main feature of concentrated disposal, also displays a number of other features and specific barriers, it was considered appropriate to perform a separate evaluation of them.

Inspection and Monitoring

The 100 years of active cooling of the WP-Cave can be regarded as a period of enhanced possibilities to retrieve the fuel and to monitor and inspect the repository.

As long as the spent fuel is stored at CLAB it is fully accessible. Irrespective of the disposal concept, the disposal steps after encapsulation will progressively increase the cost of retrieval. The cost will depend on how encapsulation is carried out and the degree of backfilling and sealing of the repository. In a repository in crystalline rock, the waste is always retrievable as long as the canister has not been penetrated.

The safety value of an extended period with a high level of retrievability is in principle very doubtful, since it often can only be achieved by delaying the completion of the passive long-term barriers and the sealing procedure.

Based on the above considerations, the group believes that the spent fuel should be kept in CLAB and no encapsulation initiated until all reasonable doubt as to the safety of the disposal concept has been dispelled. Furthermore, the group believes that the probability of retrieval being necessary for reasons of safety can be made very small by applying a multi-barrier principle and appropriate diversity and redundancy in the safety functions of the barriers. For such repository concepts the cost difference for retrieval of waste during various phases will not be of major importance.

An extended period of inspection and monitoring can provide better data for the final safety assessment prior to sealing for any repository. Based on the performance assessments conducted, the group concludes, however, that the major uncertainties in both the WP-Cave and KBS-3 concepts are in areas that could benefit from the information obtained from such an in situ measurement period.

The group concludes that the option of an intermediate retrievability period of 100 years duration does not constitute a significant advantage compared to the KBS-3 system. The required period of active cooling is considered to be a disadvantage.

Predetermined and Fixed Geometry

The active cooling of the WP-Cave and the concentrated disposal requires a predetermined and a fixed geometry of the repository.

Compared with WP-Cave, the KBS-3 system allows some flexibility in selecting the length and direction of its tunnels during excavation, or in selecting the exact positions of the canister holes in the deposition tunnels. This flexibility can be used to avoid both vertical and horizontal fissure zones in the bedrock and to avoid using volumes of low quality rock for emplacement.

To avoid less suitable rock, the WP-Cave concept must depend to a higher degree on the site investigations prior to site selection. Depending of the frequency of unacceptable rock quality or fissure zones, this might pose a problem in the siting of the WP-Cave.

The group concludes that the fixed geometry of the WP-Cave will make it more difficult to utilize the full potential of the bedrock as a natural barrier, although the drilling of the hydraulic cage will probably yield a more accurate knowledge of the rock surrounding the repository.

Elevated Temperature

The concentrated disposal gives rise to a substantially elevated temperature in the repository.

If the maximum temperature during leaching of the fuel is limited to around 150°C the group considers it possible to evaluate the safety of the repository with reasonable certainty.

However, the elevated temperature has a very direct impact on

– the flow of water through the repository, affecting nuclide release,
– the solubilities of the radionuclides, resulting in greater uncertainties in the higher temperature region,

– the increased solubility of silica with temperature, giving rise to an uncertainty in the long-term stability of the conductivity around the hydraulic cage and in the sand-bentonite barrier.

The group finds that the higher temperature in general gives rise to a higher uncertainty in the performance assessments.

Barrier System

There are also significant differences in the barrier system in the two concepts.

The performance assessment clearly demonstrates the dominant role of the bentonite barrier in the total WP-Cave-barrier system.

Evaluating the effect of the hydraulic cage is difficult due to insufficient knowledge of the channeling of the groundwater flow and the long-term stability of the function of the cage.

As a consequence of the difference between the canister materials selected for WP-Cave (steel) and for KBS-3 (copper), the steel canister will start leaking after only a couple of hundred years. On the other hand, the large amounts of iron and iron corrosion products lend high confidence to the prediction that the redox front will stay close to or inside the canister.

One major factor of uncertainty with iron in the repository system is associated with the possibility of large amounts of hydrogen, produced by anaerobic corrosion of the mild steel. There must be confidence in the capability of the bentonite layer and the surrounding bedrock to vent the gases before high pressures build up inside the buffer.

The group concludes that there is a basic similarity in the potential of the barrier systems of KBS-3 and WP-Cave. However, the selected materials, dimensions etc., that the barrier system is dominated by the sand-bentonite layer in the WP-Cave are such, repository. The WP-Cave repository will consequently rely very heavily on the integrity of that barrier. The redundancy of the engineered barrier system in WP-Cave is thus less and the amount of diversity in the mechanisms guaranteeing the basic safety of the repository is minimal.

The group also finds that substantial problems have been encountered in evaluating the safety value of the barriers, often due to uncertainties in modeling certain mechanisms (such as the possibility of porosity change in the hydraulic cage) or a lack of data (on the chemical situation in the near field). To reduce the uncertainties within the two areas mentioned, substantial international cooperation would be required over long periods.

In the course of the discussions within SFG it was found that a number of the negative factors in the WP-Cave barrier system could be reduced by replacing the steel canister with a copper canister. The copper canister would have a longer life, whereby fuel leaching would not coincide with the peak temperatures and there would be better redundancy and diversity in the barrier system. Furthermore, gas generation would not pose a problem. These issues have not been evaluated within this project.

4.4 COSTS

Cost calculations show a clear cost disadvantage for the WP-Cave system. A total cost of about SEK 44 billion has been calculated for a storage and disposal system with seven WPC 1100s, compared with about 28 billion for a KBS-3 system.

In these calculations, a number of variations have been investigated to ascertain the potential for cost reduction in WP-Cave. The only realistic way to bring the total cost down to the KBS-3 level would be to utilize fewer and bigger caves, and to reduce the thickness of and the quality requirements on the bentonite barrier. Both of these alternatives would, however, have a negative effect on radiological safety.

The uncertainties in the cost calculations are undoubtedly great, especially with regard to future cost. However, the similarity between the two concepts is so great that the relative uncertainty is small.

Although the costs have not been computed for optimal design, neither for the WP-Cave, nor for the KBS-3 concept, the group finds this cost difference to be of such significance that substantial safety advantages would be required in the WP-Cave design to compensate for it.

4.5 UNCERTAINTIES AND CONFIDENCE

A basic requirement in any final safety assessment used for licensing is that it can be shown that the uncertainties in the evaluation and the sensitivity of the calculated results to these uncertainties will not cause a change in the acceptability verdict.

Although this comparison between the alternatives does not require such a quantitative assessment of the uncertainties, the issue itself is of such importance for the concept of provability and confidence that a rather detailed discussion has been devoted to it in the report.

The group concludes that the generic uncertainties in the long-term performance of the repository are substantial whatever concept is selected for the repository design. The uncertainties specific to the WP-Cave concept are more pronounced than those associated with KBS-3. These uncertainties are, however, not so great as to prevent the conclusion from being drawn that the differences in performance between the WP-Cave and the KBS-3 concepts clearly favor KBS-3.

4.6 RECOMMENDATIONS

Based on conclusions reached in comparing the WP-Cave concept to the KBS-3 concept, the former was found to have a distinct economic disadvantage. No substantial safety advantages could be found in the WP-Cave concept and, although it might well be developed to acceptable safety, it was found to have a less redundant barrier system with higher uncertainties in performance. To choose the WP-Cave as the prioritized R&D direction would thus be more costly and the uncertainties for a successful outcome would be greater.

The SFG (the Integrated Performance Group) therefore recommends that SKB cease further development of WP-Cave as an integrated concept. There are, however, ideas and barrier principles within the WP-Cave concept that can also be utilized in other less concentrated repository concepts, e.g. the steel canister and the principles of a hydraulic cage. Investigations in these areas are being conducted within SKB and may well be continued. As the main option, however, the SFG recommends that SKB choose the distributed concept, along the lines of KBS-3, as the prioritized future direction of R&D.

REFERENCES

1-1 Handling and final disposal of nuclear waste. Programme for research and development and other measures.
Swedish Nuclear Fuel and Waste Management Co., September 1986

2-1 Kostnader för kärnkraftens radioaktiva restprodukter (Cost for managing the waste from nuclear power production, in Swedish).
Swedish Nuclear Fuel and Waste Management Co., June 1986

2-2 Skagius K and Svemar Ch;
Performance and safety analysis of the WP-Cave concept, in preparation

2-3 Pusch R, Ranhagen L and Nilsson K;
Laboratoriebestämning av hydraulisk konduktivitet och gaskonduktivitet samt svällningsegenskaper hos SFR-buffertar (Laboratory measurements of hydraulic conductivity, gas conductivity and swelling properties of SFR buffers, in Swedish).
SFR Report 84-05, Swedish Nuclear Fuel and Waste Management Co., 1984

3-1 Skagius K and Svemar Ch;
Performance and safety analysis of the WP-Cave concept, in preparation

3-2 Final Storage of Spent Nuclear Fuel — KBS-3.
Swedish Nuclear Fuel Supply Co., May 1983

3-3 Alternativa tidplaner för hantering av använt kärnbränsle (Alternative time schedules for management of spent nuclear fuel, in Swedish).
Swedish Nuclear Fuel and Waste Management Co., December 1985

3-4 Pusch R;
Stability of deep-sited smectite minerals in crystalline rock — chemical aspects.
KBS Technical Report TR 83-16, Swedish Nuclear Fuel Supply Co., March 1983

3-5 Fritz B, Kam M and Tardy Y;
Geochemical simulation of the evolution of granitic rocks and clay minerals submitted to a temperature increase in the vicinity of a repository for spent crystalline rock — chemical aspects.
KBS Technical Report TR 84-10, Swedish Nuclear Fuel and Waste Management Co., July 1984

3-6 Fritz B, Made B and Tardy Y;
Geochemical modelling of the evolution of a granite-concrete-water system around a repository for spent nuclear fuel, April 1988
SKB Technical Report TR 88-18

3-7 Kam M and Fritz B;
Geochemical modelling of the hydrothermal alteration due to heat storage i deep rock reservoirs.
5th International Symposium on Water-Rock Interaction, WRI V, Reykjavik, Island, August 8—17, 1986 pp 306—309

3-8 Claesson T;
Water-rock interaction at elevated temperatures: chemical changes in water composition, Doctoral theses at the Department of Geology at Chalmers Technical University, Gothenburg, 1983

3-9 Bruno J and Puigdomenèch I;
Validation of the SKBU1 uranium thermodynamic data base for its use in geochemical calculations with EQ3/6. (Submitted to the MRS Symposium in Berlin, October 1988)

3-10 Allard B, Kipatsi H and Rydberg J;
Sorption av långlivade radionuklider i lera och berg (Sorption of long-lived radionuclides in clay and rock, in Swedish).
KBS Technical Report TR 55, October 1977

3-11 Natural Fission reactors, IAEA report STI/PVB/475, Vienna, 1978

3-12 Tammemagi H Y, Haverslew B and Sturcho N C;
Investigations of the Empire Creek Stock, Montana, as an analogue to a nuclear waste repository
Chem.Geol. vol 55, 1986 pp 375—386

3-13 Gascoyne M;
The use of uranium-series disequilibrium for site characterization and as an analogue for actinide migration
CEC Symposium on Natural Analogues in Radioactive Waste Disposal, CEC Report EUR 11037 EN, Brussels, 1987

3-14 Paneix J C, Menager M T, Trotignon L and Petit J C;
Hydrothermal alteration in the Auriat granite. Analogy with radwaste disposal
CEC Symposium on Natural Analogues in Radioactive Waste Disposal, CEC Report EUR 11037 EN, Brussels, 1987

3-15 Cramer J J, Vilks P and Larocque J P A;
Near-field analog Features from the Cigar Lake uranium deposit,
CEC Symposium on Natural Analogues in Radioactive Waste Disposal,
CEC Report EUR 11037 EN, Brussels, 1987

List of SKB reports

Annual Reports

1977–78
TR 121
KBS Technical Reports 1 – 120.
Summaries. Stockholm, May 1979.

1979
TR 79–28
The KBS Annual Report 1979.
KBS Technical Reports 79-01 – 79-27.
Summaries. Stockholm, March 1980.

1980
TR 80–26
The KBS Annual Report 1980.
KBS Technical Reports 80-01 – 80-25.
Summaries. Stockholm, March 1981.

1981
TR 81–17
The KBS Annual Report 1981.
KBS Technical Reports 81-01 – 81-16.
Summaries. Stockholm, April 1982.

1982
TR 82–28
The KBS Annual Report 1982.
KBS Technical Reports 82-01 – 82-27.
Summaries. Stockholm, July 1983.

1983
TR 83–77
The KBS Annual Report 1983.
KBS Technical Reports 83-01 – 83-76
Summaries. Stockholm, June 1984.

1984
TR 85–01
Annual Research and Development Report 1984
Including Summaries of Technical Reports Issued during 1984. (Technical Reports 84-01 – 84-19)
Stockholm June 1985.

1985
TR 85-20
Annual Research and Development Report 1985
Including Summaries of Technical Reports Issued during 1985. (Technical Reports 85-01-85-19)
Stockholm May 1986.

1986
TR 86-31
SKB Annual Report 1986
Including Summaries of Technical Reports Issued during 1986
Stockholm, May 1987

1987
TR 87-33
SKB Annual Report 1987
Including Summaries of Technical Reports Issued during 1987
Stockholm, May 1988

1988
TR 88-32
SKB Annual Report 1988
Including Summaries of Technical Reports Issued during 1988
Stockholm, May 1989

Technical Reports

1989

TR 89-01
Near-distance seismological monitoring of the Lansjärv neotectonic fault region Part II: 1988
Rutger Wahlström, Sven-Olof Linder, Conny Holmqvist, Hans-Edy Mårtensson
Seismological Department, Uppsala University, Uppsala
January 1989

TR 89-02
Description of background data in SKB database GEOTAB
Ebbe Eriksson, Stefan Sehlstedt
SGAB, Luleå
February 1989

TR 89-03
Characterization of the morphology, basement rock and tectonics in Sweden
Kennert Röshoff
August 1988

TR 89-04
SKB WP-Cave Project
Radionuclide release from the near-field in a WP-Cave repository
Maria Lindgren, Kristina Skagius
Kemakta Consultants Co, Stockholm
April 1989

TR 89-05
SKB WP-Cave Project
Transport of escaping radionuclides from the WP-Cave repository to the biosphere
Luis Moreno, Sue Arve, Ivars Neretnieks
Royal Institute of Technology, Stockholm
April 1989

TR 89-06
SKB WP-Cave Project
Individual radiation doses from nuclides contained in a WP-Cave repository for spent fuel
Sture Nordlinder, Ulla Bergström
Studsvik Nuclear, Studsvik
April 1989

TR 89-07
SKB WP-Cave Project
Some Notes on Technical Issues
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Stefan Björklund, Lennart Josefson
Division of Solid Mechanics, Chalmers University of Technology, Gothenburg, Sweden
Part 2: Gas and water transport from WP-Cave repository Luis Moreno, Ivars Neretnieks
Department of Chemical Engineering, Royal Institute of Technology, Stockholm, Sweden
Part 3: Transport of escaping nuclides from the WP-Cave repository to the biosphere. Influence of the hydraulic cage
Luis Moreno, Ivars Neretnieks
Department of Chemical Engineering, Royal Institute of Technology, Stockholm, Sweden
August 1989

TR 89-08
SKB WP-Cave Project
Thermally incuded convective motion in groundwater in the near field of the WP-Cave after filling and closure
Polydynamics Limited, Zürich
April 1989

TR 89-09
An evaluation of tracer tests performed at Studsvik
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[1] The Royal Institute of Technology, Department of Chemical Engineering, Stockholm
[2] Studsvik Nuclear, Nyköping
March 1989

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Lars B Ekbom, Sven Bogegård
Swedish National Defence Research Establishment
Materials department, Stockholm
February 1989

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[1] Swedish Geological Co, Uppsala
[2] EMX-system AB, Luleå
February 1988

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Jan-Erik Andersson, Lennart Ekman, Erik Gustafsson, Rune Nordqvist, Sven Tirén
Swedish Geological Co, Division of Engineering Geology
June 1988

TR 89-13
Spent fuel
Dissolution and oxidation
An evaluation of literature data
Bernd Grambow
Hanh-Meitner-Institut, Berlin
March 1989

TR 89-14
The SKB spent fuel corrosion program
Status report 1988
Lars O Werme[1], Roy S Forsyth[2]
[1] SKB, Stockholm
[2] Studsvik AB, Nyköping
May 1989

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Comparison between radar data and geophysical, geological and hydrological borehole parameters by multivariate analysis of data
Serje Carlsten, Lennart Lindqvist, Olle Olsson
Swedish Geological Company, Uppsala
March 1989

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Documentation of the 1000 m hydrofracturing unit at Luleå University of Technology
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[1] Abem AB, Uppsala, Sweden
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June 1989

TR 89-19
Characterization of fracture zone 2, Finnsjön study-site
Editors: K. Ahlbom, J.A.T. Smellie, Swedish Geological Co, Uppsala
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K. Ahlbom and J.A.T. Smellie. Swedish Geological Company, Uppsala, Sweden.
Part 2: Geological setting and deformation history of a low angle fracture zone at Finnsjön, Sweden
Sven A. Tirén. Swedish Geological Company, Uppsala, Sweden.
Part 3: Hydraulic testing and modelling of a low-angle fracture zone at Finnsjön, Sweden
J-E. Andersson[1], L. Ekman[1], R. Nordqvist[1] and A. Winberg[2]
[1] Swedish Geological Company, Uppsala, Sweden
[2] Swedish Geological Company, Göteborg, Sweden
Part 4: Groundwater flow conditions in a low angle fracture zone at Finnsjön, Sweden
E. Gustafsson and P. Andersson. Swedish Geological Company, Uppsala, Sweden
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J.A.T. Smellie[1] and P. Wikberg[2]
[1] Swedish Geological Company, Uppsala, Sweden
[2] Swedish Nuclear Fuel and Waste Management Company, Stockholm, Sweden
Part 6: Effects of gas-lift pumping on hydraulic borehole conditions at Finnsjön, Sweden
J-E- Andersson, P. Andersson and E. Gustafsson. Swedish Geological Company, Uppsala, Sweden
August 1989