CONCEPT DEVELOPMENT FOR SALTSTONE

AND LOW LEVEL WASTE DISPOSAL

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ABSTRACT

A low-level alkaline salt solution will be a byproduct in the processing of high-level waste at the Savannah River Plant (SRP). This solution will be incorporated into a cement wasteform, saltstone, and placed in surface vaults. Laboratory and field testing and mathematical modeling have demonstrated the predictability of contaminant release from cement wasteforms. Saltstone disposal in surface vaults will meet drinking water standards in shallow groundwater at the disposal area boundary. Planning for new Low-Level Waste (LLW) disposal could incorporate concepts developed for saltstone disposal.

INTRODUCTION

High-level nuclear waste will be solidified in a facility now being built at the SRP near Aiken, South Carolina. This Defense Waste Processing Facility (DWPF) is scheduled to begin operations in 1989. The DWPF will solidify high-level waste, now being stored in waste tanks at SRP, by vitrification into borosilicate glass. The vitrified waste will be shipped to a federal repository(1,2).

Before vitrification, waste will be processed in the existing waste tanks. During waste processing, about 25 million gallons of soluble salts will be decontaminated. The resulting low-level decontaminated salt solution will be solidified with a cement/fly ash mixture and disposed of in surface vaults at SRP. Disposal area design is based on the results of analytical and numerical models of the release of materials from saltstone. The performance objective is to permanently dispose of waste salt in an environmentally sound manner that will meet state groundwater quality standards at the disposal area boundary(3).

LLW disposal technologies are being considered that would provide greater groundwater protection than the current disposal method of shallow land burial (SLB). The systems approach used in the saltstone program could be used in the development of the new technologies.

SALT WASTE AND WASTEFORM CHARACTERISTICS

Most of the radioactivity will be removed from high-level waste during in-tank processing by addition of sodium tetraphenylborate to precipitate cesium, and by removing strontium through adsorption onto sodium titanate. The composition of the resulting decontaminated salt solution is shown in Table I. The waste salt solution is a low-level radioactive waste (the radioactivity is about 230 µCi/liter). The salt solution is also a hazardous waste by characteristic. The waste is corrosive (pH>12.5) and EP-toxic (chromium = 160 ppm).

TABLE I

COMPOSITION OF DWPF SALT SOLUTION

Nonradioactive Component	Grams/Liter	Radioactive Component	µCi/Liter
Na	117.	^{90}Sr	0.9
NO_3^-	130.	^{99}Tc	75.3
NO_2^-	30.	^{137}Cs	24.6
OH^-	20.	Total alpha emitters	0.2
Cr	0.2 (160 ppm)		
Hg	1.5×10^{-6} (0.01 ppm)	Total Radioactivity	230
Ag	1.1×10^{-7} (0.0008 ppm)		

pH ~ 14
Density 1.3 gm/cc.

Although the waste salt solution is hazardous, the resulting saltstone is non-hazardous. Extraction Procedure-Toxicity Testing has shown that chromium and other metals in the leachate are at acceptable levels. Thus, saltstone can be disposed of as a non-hazardous, low-level radioactive solid waste.

The waste salt solution will be transferred by pipeline from the high-level waste tanks where it is decontaminated to a low-level waste processing area (Z-Area). At the waste processing area, the salt solution will be mixed with cement and fly ash and pumped to surface disposal vaults. The formulation to be used is shown in Table II.

TABLE II

BLENDED CEMENT FORMULATION, WT %

Portland Cement (Class H)	11.5	57.5 wt % cement
Fly ash (Class C)	46.0	
Salt	12.3	42.6 wt % solution
Water	30.2	

SALTSTONE LEACHING

Laboratory leaching tests have been done on samples of saltstone to define the rate of removal of materials from saltstone to the environment. Leaching tests were done by a proposed ANSI procedure(4). Cylindrical samples of saltstone were leached in deionized water. The water was changed daily for the first week, weekly for two months, and then every two weeks until the end of the test. Leachate was analyzed by ion chromatography and gamma spectrometry. Typical leaching results are shown in Fig. 1 for nitrate, technetium, cesium, and strontium. The leaching curves show a rapid decrease in leach rate between the first and second day. This is due to dissolution of salts from the sample surface. After the second day, the leach rates for nitrate, technetium, and cesium decrease in proportion to the square root of time. This behavior is indicative of a diffusion-controlled release. The leaching of strontium, however, is indicative of a dissolution controlled mechanism. Strontium probably substitutes for calcium as the cement matrix forms.

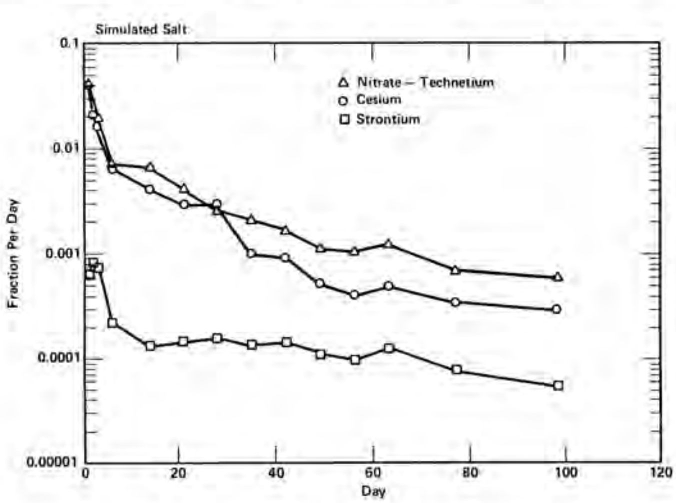

Fig. 1. Saltstone Leach Rates, Saturated Conditions.

Nitrate leach results for several saltstone samples, prepared in the laboratory and during a large-scale field test, have been analyzed to estimate the effective diffusion coefficient for saltstone. The best estimate is 5×10^{-9} cm^2/sec.

Leaching tests have also been carried out in unsaturated soil to simulate burial of saltstone in earthen trenches. These tests have been done in the same manner as the deionized water leaching tests, except that unsaturated soils surrounds the saltstone sample instead of water. Typical results of leaching in unsaturated soil are shown in Fig. 2 in comparison with results of leaching in water. In this test, the soil was at natural field water capacity, about 20% (volumetric) water. In other tests, the waste content of the soil was reduced. No effect on leach rate was observed until the soil water content was reduced to ≈1%. Figure 3 illustrates saltstone leaching in water and unsaturated soil. In either case the diffusion of salts from saltstone into the surrounding medium is the controlling mechanism and the observed leach rate is expected to be the same.

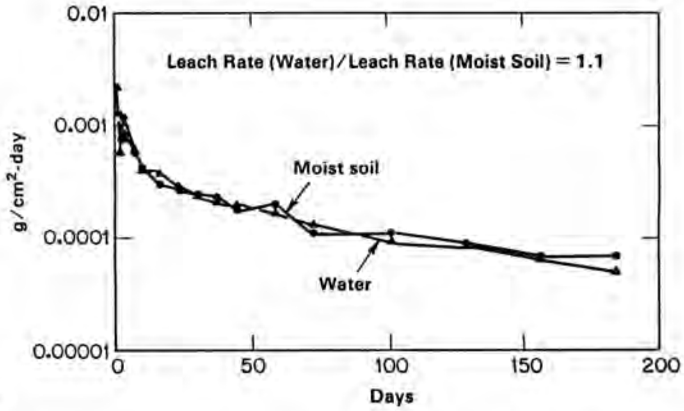

Fig. 2. Saltstone Leach Rates, Saturated vs Unsaturated Conditions

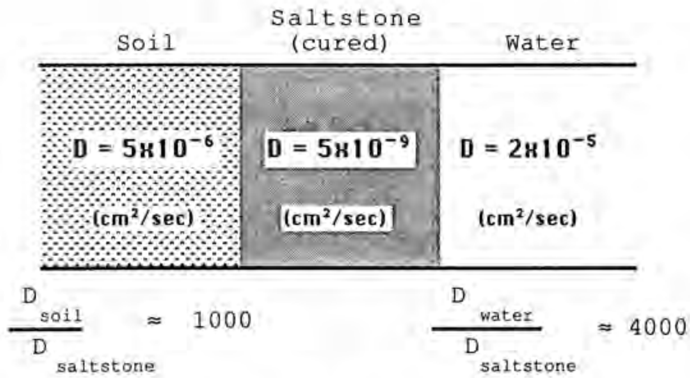

Fig. 3. Diffusion Through Cured Saltstone Controls Release.

FIELD TESTING OF SALTSTONE RELEASE

In 1983, three lysimeters were constructed to test the performance of saltstone disposal in earthen trenches. Thirty-ton monoliths of saltstone were prepared from actual, decontaminated SRP waste salt solution. Each monolith was formed by pouring saltstone grout into an earthen trench contained in a Hypalon®-lined basin. Figure 4 shows the details of construction of the lysimeters. One of the lysimeters contains only a saltstone monolith. Another lysimeter contains a saltstone monolith covered by a clay cap to divert infiltrating water. The third lysimeter contains a monolith covered by a gravel cap. Samples of percolate water have been collected from the lysimeter sumps and analyzed for nitrate and other materials since the lysimeters were installed. Soil moisture samplers were installed within six inches of the monoliths prior to pouring of the saltstone. Samples of soil moisture have been collected periodically from the lysimeters and also analyzed.

Nitrate concentrations from the three lysimeter sumps are shown in Fig. 5. Nitrate in the two capped lysimeters is coincident with natural levels of nitrate in rain and groundwater; thus, no nitrate has yet migrated to the sumps of the capped lysimeters. The uncapped lysimeter has released significant levels of nitrate to the sump. The maximum concentration to date is 209 ppm. Technetium-99 has also been observed in the uncapped lysimeter (maximum concentration of 12,000 pCi/L), but not in the capped lysimeters.

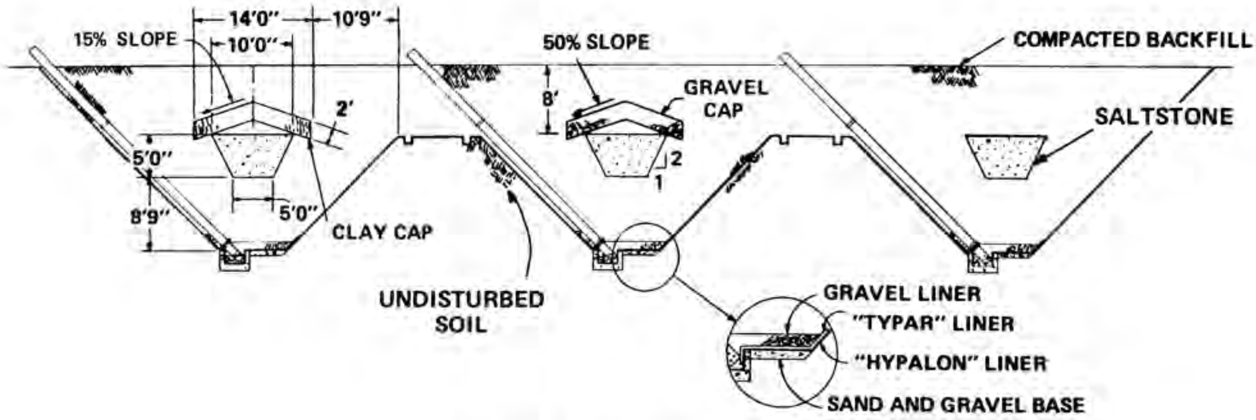

Fig. 4. Saltstone Lysimeters.

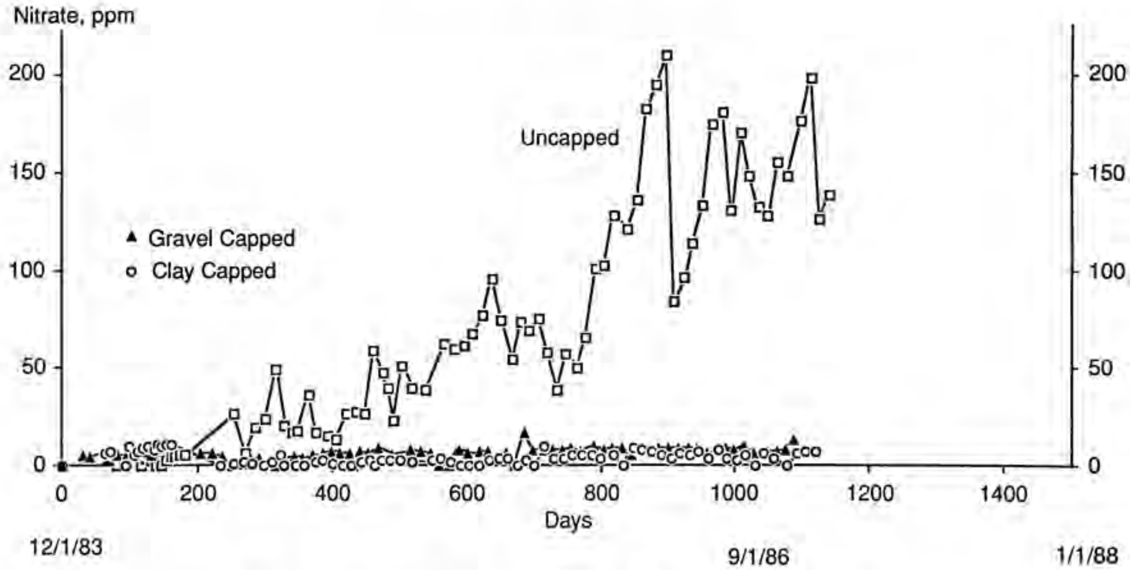

Fig. 5. Saltstone Lysimeters Nitrate Release to Sump.

Results of soil moisture samples show significant concentrations of nitrate (up to 15,000 ppm) and technetium-99 (up to 450,000 pCi/L) adjacent to the two capped lysimeters. This demonstrates that soluble components are released from the saltstone, but that the caps are effective in reducing the amount of water flowing next to the monolith, which could carry the materials to the sump.

SALTSTONE RELEASE MODELING

A performance goal for the disposal of saltstone is maintaining groundwater quality standards at the disposal area boundary. The State of South Carolina has adopted the EPA drinking water standards as the standard to be applied to groundwater similar to that underlying the saltstone disposal area. Mathematical modeling (both analytic and numeric) was applied to predict releases of nitrate and other materials from various saltstone disposal designs.

Numeric models were used at the Savannah River Laboratory (SRL) and by a subcontractor (Intera Technologies, Austin, Texas). Numeric models were validated by applying them to the saltstone lysimeters and comparing model predictions with lysimeter observations.

Results of model validation are shown in Fig. 6. The SRL model assumed that water moved through the soil as if the soil were saturated with water (potential flow)(5). The Intera model used measured rainfall at the lysimeter site and measured unsaturated flow properties of soil and saltstone in a full two-phase, unsaturated model to simulate the movement of water through unsaturated porous media.

SALTSTONE DISPOSAL AREA DESIGN

A saltstone disposal design has been developed that has been effective in maintaining groundwater quality. The design, shown in Fig. 7, is a surface disposal vault constructed of concrete and containing about one year's production (56,000 cubic yards) of saltstone. The floor is 2.5 feet thick, the sidewalls are 1.5 feet thick and the sloped concrete top is 2 feet thick. After the vaults are closed, the disposal site will be decommissioned. A potential decommissioning concept involving mounding with earth and installing a clay cap over the vault is also shown in Fig. 7. The validated Intera model has been applied to predict nitrate concentrations in groundwater resulting from the decommissioned vault. Predicted nitrate concentrations are shown in Fig. 8. All contaminants are predicted to be below groundwater standards at the disposal area boundary.

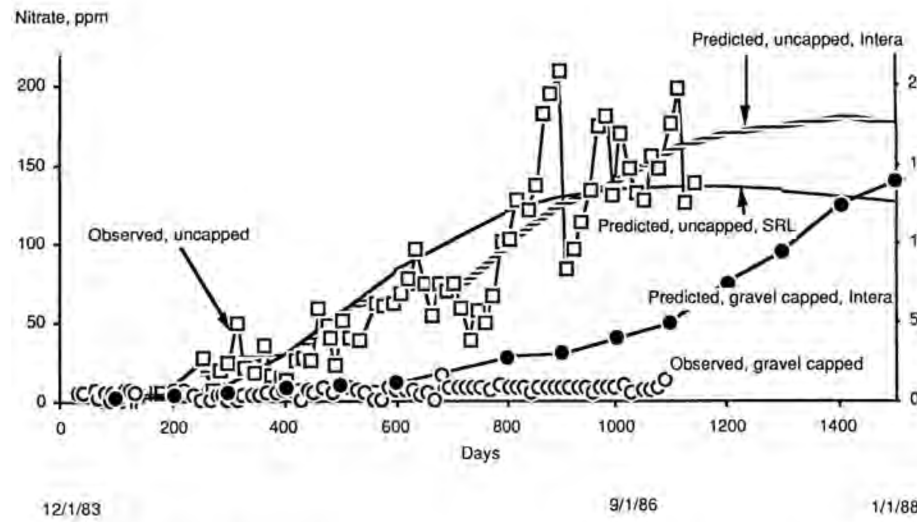

Fig. 6. Gravel Capped and Uncapped Saltstone Lysimeter SRL and Intera Model Results.

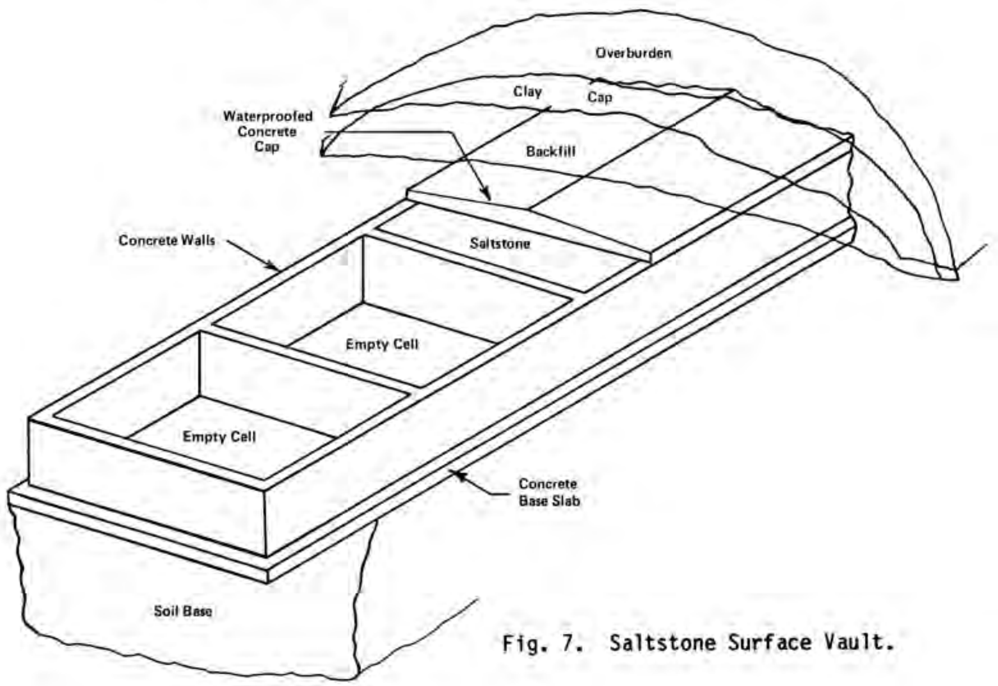

Fig. 7. Saltstone Surface Vault.

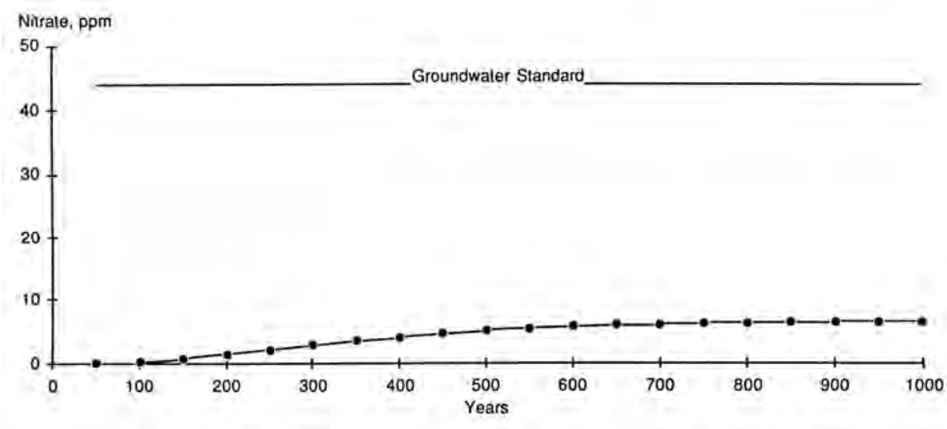

Fig. 8. Nitrate in Groundwater from Saltstone Vault Disposal.

LOW-LEVEL RADIOACTIVE WASTE DISPOSAL CONCEPT

The same principles that were developed for saltstone disposal at SRP can be applied to the design of disposal facilities for low-level radioactive waste. The disposal of radioactive waste in a manner that prevents the accumulation of harmful concentrations in the environment is facilitated by radioactive decay. The delay factors introduced by wasteform liners and clay caps have a large protective effect toward the shallow groundwater. The dominant mechanism for radionuclide release is diffusion if the waste is fixed in concrete. The performance improvement of a concrete wasteform compared to waste buried directly in soil is based on the differences in diffusivity. The diffusivity of soluble species such as nitrate in soil is estimated to be about 5×10^{-6} cm²/sec; in a concrete wasteform, the diffusivity is about 5×10^{-8} cm²/sec. Containment factors from wasteforms and liner for a vault system for LLW are summarized in Table III. The overall combined effect of wasteforms, liner, and cap with decay is summarized in Fig. 9.

TABLE III

CONTAINMENT FACTORS FOR ENGINEERED BARRERS (NO DECAY)

• Concrete Waste Form	
Diffusion	10
Solubility + Diffusion	5
• Concrete Liner	10
• Total Factor	100 to 500

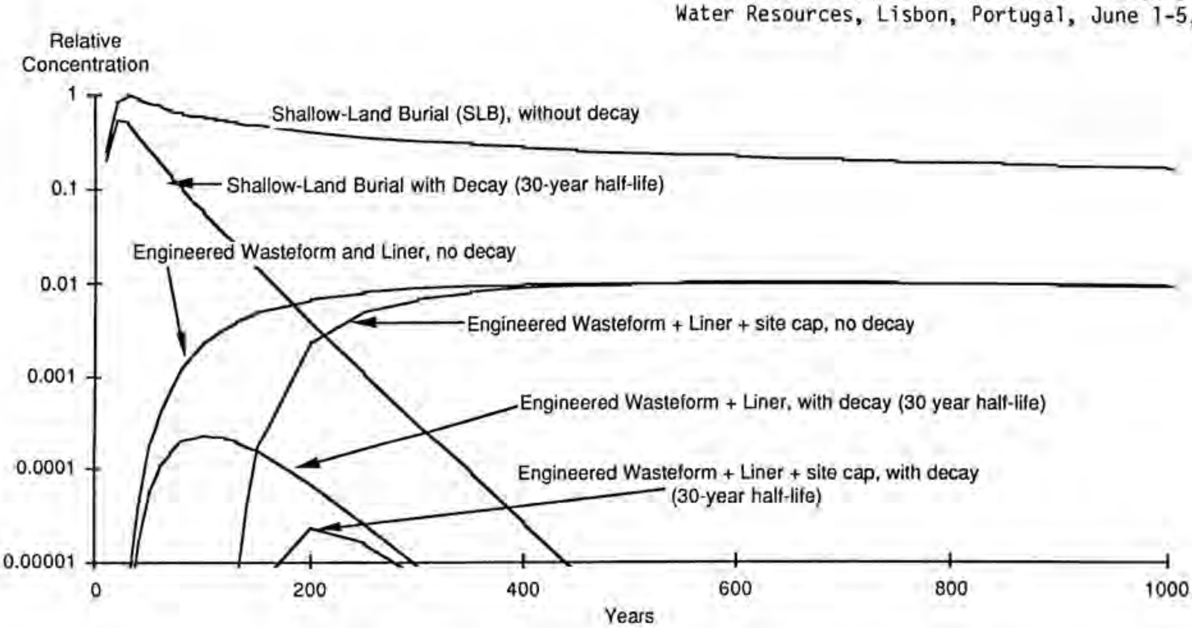

Fig. 9. Effect on Groundwater Concentrations Engineered Wasteform + Liner and Site Cap.

In summary, release of radioactivity from a LLW vault is projected to be extremely low for over 100 years. Peak concentrations in the groundwater is expected after about 200 years. This performance is better than that of shallow land burial by a factor greater than 10^4.

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