December 8, 1998

FOR: The Commissioners

FROM: William D. Travers /s/
Executive Director for Operations

SUBJECT: TRANSFERS OF MATERIAL CONTAINING LESS THAN 0.05 PERCENT BY WEIGHT SOURCE MATERIAL UNDER 10 CFR 40.51(b)(3) AND (b)(4), AND 40.13(a)

PURPOSE:

To request Commission approval, by negative consent, of the staff's plans to approve a licensee's request to transfer baghouse dust containing less than 0.05 percent by weight (wt%) source material to unlicensed persons in accordance with 10 CFR 40.51(b)(3) and (b)(4), and 40.13(a); and to approve a similar request for transfer of slag, if the projected dose is determined to be less than 1 mSv/yr (100 mrem/yr).

BACKGROUND:

The Shieldalloy Metallurgical Corporation (SMC), has two separate U.S. Nuclear Regulatory Commission (NRC) licensed sites, one located in Newfield, New Jersey, and the other in Cambridge, Ohio. The SMC site in Newfield, New Jersey, produces baghouse dust contaminated with uranium and thorium, during production of specialty alloys for the steel industry. By letter dated June 24, 1996, SMC, Newfield, requested authorization to transfer baghouse dust containing less than 0.05 wt% source material to an exempt person, in accordance with 10 CFR 40.51(b)(3) and 40.13(a). In addition, an SMC contractor sent a letter to NRC dated November 19, 1997, which concluded that radioactive slag at SMC's Cambridge facility could similarly be transferred and that the doses would be a very small fraction of the 1 Sv/yr (100 mrem/yr) public dose limit in 10 CFR Part 20. Finally, in a May 13, 1998, letter to NRC, SMC Newfield described its plans and justification for releasing radioactive slag from its facility to unlicensed persons. The Newfield baghouse dust would be transferred to a cement manufacturer as a substitute for lime in cement; the slag would be furnished to steel manufacturers for steel production. These different proposals are similar in a number of respects. Each is concerned with byproducts of the company's specialty alloy production that contain less than 0.05 wt% source material; each byproduct has a further beneficial use when transferred to an unlicensed "person" such as a steel refinery or cement manufacturer; and each raises policy and potential safety issues.

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The regulations in 10 CFR 40.51(b)(3) and 40.13(a) allow licensees to transfer source material to any person exempt from the licensing requirements of the Atomic Energy Act (AEA) and 10 CFR Part 40, as long as the source material content is less than 0.05 wt% of the material as a whole. However, under some circumstances, transfer of material in accordance with these regulations could result in doses that exceed the 1 mSv/yr (100 mrem/yr) public dose limit contained in 10 CFR 20.1301(a)(1). In addition, because the regulations do not explicitly provide a regulatory basis for denying such transfers, and because a licensee making such transfers would be complying with the regulations in Part 40 as they are currently written, the NRC may issue an order to stop a licensee from making such a transfer only when the transfer may result in a potentially hazardous condition that could affect public health and safety. Because of the potential health risks associated with these transfers, the staff reviewed SMC's requests and performed several site-specific dose assessments. Based on those assessments, the staff has concluded that although a licensee would be in compliance with the regulations in Part 40, doses could occur which are in conflict with the public dose limits contained in 10 CFR 20.1301(a)(1), and that allowing these transfers could set a precedent without resolving that conflict. For this reason, the staff developed this paper.

DISCUSSION:

Section 40.51(b)(3) allows a licensee to transfer an unlimited quantity of source material to any person exempt from the licensing requirements of the AEA and regulations in Part 40; Section 40.51(b)(4) allows transfers to persons exempt from licensing under Agreement State regulations. Section 40.13(a) states that any person is exempt from the regulations in Part 40 and the requirements for a license set forth in Section 62 of the AEA, to the extent that such person receives, possesses, uses, transfers, or delivers source material in any mixture, compound, solution, or alloy in which the source material is less than 0.05 wt% of the mixture, compound, solution, or alloy. Based on the staff's review of the supporting record for Part 40, it is not possible to determine to what extent, if any, public health and safety were considered in promulgating 10 CFR 40.13(a).(1)

It is important to note that, because 10 CFR 40.13(a) exempts persons receiving this material from licensing under the AEA, these persons cannot be held to any of the requirements for licensing in Part 40 or the requirements for radiological protection, waste disposal, or decommissioning, including the radiological criteria for license termination in 10 CFR Part 20, Subpart E. Once material is transferred out of the licensee's control, NRC can no longer place restrictions on the use of the material to reduce potential doses, absent an NRC order based on health and safety.

Transfer of material, under Sections 40.51(b) and 40.13(a), may be inconsistent with the radionuclide concentrations and dose limits allowed in past NRC actions for releasing contaminated sites and with future actions under the final rule on the radiological criteria for license termination. The 0.05 wt% translates to approximately 12.5 Bq/g (339 pCi/g) for natural uranium (including U-238, 235, and 234, and omitting consideration of decay products) or 4.29 Bq/g (116 pCi/g) for natural thorium (Th-228 and 232). These concentrations of uranium and thorium are 34 times and 12 times, respectively, greater than the levels NRC has typically used for releasing contaminated sites (less than or equal to 0.37 Bq/g (10 pCi/g) natural uranium; 0.37 Bq/g (10 pCi/g) natural thorium, with decay products present in equilibrium) under Option 1 of the 1981 Branch Technical Position entitled, "Disposal or Storage of Thorium or Uranium Wastes from Past Operations." Depending on the scenario assumed and the methodology used to calculate the dose,

these concentrations could result in doses much greater than the 0.25 mSv/yr (25 mrem/yr) dose limit for unrestricted release contained in the final rule on radiological criteria for license termination. In addition, if radium is in equilibrium with the uranium or thorium parent in the chain, the radium activities would significantly exceed the soil standard in Criterion 6, Appendix A, 10 CFR Part 40, for release of sites for unrestricted use.

Although permitting these types of transfers could lead to doses, in certain situations, that exceed NRC's public dose limits, it should be noted that the concentrations of uranium and thorium that can be transferred under 10 CFR 40.51(b)(3) and (b)(4), although defined as source material, are similar to the concentrations found in some types of naturally occurring radioactive material (NORM). The volume of NORM waste produced every year in the U.S.

[1 billion metric tons per year (1.1 billion short tons per year)] is far greater than the volume of baghouse dust SMC is requesting to transfer [12,000 metric tons (13,000 short tons)], and the doses to members of the public from NORM could be substantially higher in some circumstances than those estimated for the SMC baghouse dust. Note, however, that much of NORM waste, such as fly ash, has substantially lower radionuclide concentrations than the baghouse dust.

The Newfield Site. SMC, Newfield, proposed to ship its onsite inventory of baghouse dust [15,000 m^3 ($5.3x10^5$ ft^3)] to a cement manufacturer, to be used as a substitute for lime in the production of cement. NRC calculated doses to cement workers located outside the licensee's facility by using two methodologies: (1) the dose methodology of the International Commission on Radiological Protection (ICRP) Publications 26 [1977] and 48 [1986], which is consistent with 10 CFR Part 20, and (2) the new methodology of ICRP Publications 66 [1994] and 68 [1994].(2) Input parameters the licensee submitted were not necessarily relied on, because once the baghouse dust is transferred, the licensee no longer has control over this material. As discussed in Attachment 1, the dose assessment the staff performed estimated that the highest expected total effective dose equivalent to the maximally exposed individual could be on the order of 1.30 mSv/yr (130 mrem/yr), based on ICRP Publications 26 and 48. Using the new methodology of ICRP Publications 66 and 68, the staff determined that the dose would not exceed 0.35 mSv/yr (35 mrem/yr) and is expected to be as low as 0.02 mSv/yr (2 mrem/yr). Note also that the State of New Jersey has been informed of the staff's review.

Based on the results of these analyses, and the conservatism used in the calculation, it is unlikely that a cement worker would be exposed to the maximum value calculated by either method. The staff has concluded that the transfer of the SMC, Newfield, site baghouse dust would be less than 100 mrem/yr and would subsequently not cause a significant health hazard. The staff will send a letter to the licensee indicating that NRC does not object to the transfer (see Attachment 2). It should be noted that although the transfer is acceptable in this case, this could be a significant issue for the shipment of slag from SMC, Cambridge site, as well as other sites in the future because, depending on the material, end use, and dose scenario, there may be a potential for public doses to exceed 1 mSv/yr (100 mrem/yr) for some transfers. As mentioned earlier in this paper, this application of the regulations in Part 40 could result in a dose which is in direct conflict with the public dose limit contained in Part 20.

The Cambridge Site. The slag at the Cambridge site has a different composition than the baghouse dust at Newfield and contains a significant concentration of thorium-230. NRC staff, including Office of Nuclear Regulatory Research staff and its contractor, Sciences Applications International Corporation (SAIC EXIT), have reviewed the SMC dose assessment provided by its contractor in the November 19, 1997, letter and independently performed a variety of dose assessments for the Cambridge slag that SMC proposes to ship to steel manufacturers. Note also that the staff coordinated its review with the State of Ohio.

SMC calculated that the doses associated with the processing of the slag would be approximately 1 mrem/yr (0.01 mSv/yr). The staff's contractor evaluation of the potential doses estimated a much higher potential dose, on the order of 1 rem/yr (0.01 Sv/yr), using more conservative assumptions, which is well in excess of the 10 CFR Part 20 limit. The limiting scenario in either case is for a group of workers (at a steel refinery, slag blender, or disposal site), whose primary job is using heavy earthmoving equipment to move, load, and unload slag. The slag is stored in a pile outdoors, and the workers are exposed to contamination via external inhalation and secondary ingestion pathways. The primary exposure pathway is dust inhalation. SMC did not consider this exposure scenario, but instead, for plant workers, considered a brief exposure to a dropped load of slag and exposure to direct gamma radiation.

To reconcile the differences between SMC and NRC results, the staff is working with SMC, Cambridge, to determine realistic and appropriate user scenarios and input parameters for the dose assessment. By letter dated October 13, 1998, NRC forwarded a copy of the SAIC assessment and requested a meeting with SMC to resolve these issues. If, after meeting with SMC to reconcile differences in the dose assessments, the projected dose is determined to be less than 1 mSv/yr (100 mrem/yr), the staff will send a letter to the licensee indicating that NRC does not object to the transfer. If the projected dose is determined to be greater than 1 mSv/yr (100 mrem/yr), the Commission will be notified before any action is taken.

In addition to the above site-specific actions, the staff is currently working on developing a range of possible revisions to Part 40, as described in the semi-annual "Rulemaking Activity Plan," to help resolve this issue, as well as several other issues involved with Part 40, through a contractor report entitled "Options Paper on Exemption in 10 CFR Part 40 for <0.05% Source Material" (options paper). This options paper outlines several of the generic issues associated with Part 40, and the need for revisions, and discusses and evaluates legislative and rulemaking options to resolve these issues. Staff expects to receive the final draft of the report during the first half of December 1998.

Although the staff is considering future modifications to Part 40 or possible legislation because of concerns with the 0.05 wt% threshold, the Commission should be aware that some licensees are actively considering using 10 CFR 40.51(b)(3) and (b)(4) to transfer material containing less than 0.05 wt% source material to unlicensed persons. In addition, SMC could currently ship the baghouse dust from the Newfield site or the slag from its Cambridge or Newfield sites, under Part 40, in accordance with NRC regulations. As mentioned earlier in this paper, this application of the regulations in Part 40 could result in a dose which is in direct conflict with the public dose limit contained in 10 CFR 20.1301(a)(1), since there is no assurance that all doses as a result of these potential transfers allowed by Part 40 would be consistent with NRC radiation protection standards set forth in Part 20.

RECOMMENDATIONS:

The staff intends to approve SMC's request to transfer baghouse dust from the Newfield, New Jersey site to an exempt person. In addition, the staff intends to approve SMC's request to transfer the slag from the Cambridge, Ohio site to an exempt person, if the projected dose is determined by the staff to be less than 1 mSv/yr (100 mrem/yr). If the projected dose is determined to be greater than 1 mSv/yr (100 mrem/yr), the Commission will be notified before any action is taken.

Staff requests action within 10 days. Action will not be taken until the Staff Requirements Memorandum is received. We consider this action to be within the delegated authority of the Director, Office of Nuclear Material Safety and Safeguards.

COORDINATION:

OGC has reviewed this paper and has no legal objection. The Office of the Chief Information Officer has reviewed the Commission Paper for information technology and information management implications and concurs in it.

original /s/ by
William D. Travers
Executive Director for Operations

Attachments: As stated

ATTACHMENT 1

NRC DOSE ASSESSMENT FOR
SHIELDALLOY METALLURGICAL CORPORATION
BAGHOUSE DUST FROM NEWFIELD, NJ PLANT

The Shieldalloy Metallurgical Corporation (SMC), at its site in Newfield, New Jersey, processes pyrochlore, a concentrated ore containing columbium (also known as niobium), to produce specialty alloys, slag fluidizers, and other products for the steel industry. Production of specialty alloys at the site produces baghouse dust that is contaminated with uranium (U) and thorium (Th) originally present in the pyrochlore. By letter dated June 24, 1996, SMC requested authorization to transfer baghouse dust containing less than 0.05 percent by weight (wt%) source material to an exempt person, in accordance with 10 CFR 40.51(b)(3) and 40.13(a).

When SMC made a request to transfer its onsite inventory of baghouse dust [15,000 m^3 ($5.3x10^5$ ft^3)] to a cement manufacturer to be used as a substitute for lime in the production of cement, it performed a dose assessment, considering a limited range of exposure scenarios, and determined that the individual doses as a result of this transfer would be small [e.g., less than 0.05 mSv/yr (5 mrem/yr)].

The staff performed an independent dose assessment and concluded that the individual at the cement manufacturing plant whose primary job responsibilities include using heavy earthmoving equipment to move, load, and unload baghouse dust would be expected to receive the highest dose from use of this material. The staff assumed that the baghouse dust would be stored in a pile outdoors adjacent to the plant and that the worker would be exposed to contamination via external and inhalation pathways.

The staff calculated doses using two different methodologies: (1) that of the International Commission on Radiological Protection (ICRP) Publications 26 [1977] and 48 [1986], which is consistent with 10 CFR Part 20, and (2) that of ICRP Publications 66 [1994] and 68 [1994]. The input parameters to the dose assessment, including the source term and exposure times, are discussed below.

The staff assumed for assessment purposes that the baghouse dust could contain up to 0.05 wt% uranium and thorium. Although the licensee's dose assessment assumed that the concentration of uranium and thorium in the baghouse dust is 0.03 wt%, the licensee has committed only to ensuring that the baghouse dust contains concentrations of uranium and thorium of less than 0.05 wt%. In addition, baghouse dust containing uranium and thorium at or exceeding the 0.05 wt% concentration limit exists at the site [Smith, 1998]. In addition, the staff assumed a Th/U activity ratio of 5.37 based on SMC submittals dated June 24, 1996, June 5, 1997, and May 13, 1998.(3) Therefore, the staff assumed a U-238 concentration of 0.42 Bq/g (11.4 pCi/g) and a Th-228 concentration of 1.9 Bq/g (51.4 pCi/g). Note that the decay products of uranium and thorium were considered to be in secular equilibrium with parent radionuclides. However, the dose due to radon was not included in the assessment because, in the limiting scenario, exposure takes place outside where radon concentrations would not expected to be significant.

The staff also estimated the amount of dust in the air of a respirable size. The licensee proposed a dust loading factor of 0.2 mg/m^3. This value is cited in NCRP Publication 77 [1984], which states that this is an appropriate dust loading factor for outdoor work in dusty occupations. This is also the default value used in the computer modeling code RESRAD. The RESRAD manual [Yu et al., 1993] indicates that this value has been suggested by Gilbert, et al. [1983] as an average factor which takes into account short periods of high mass loading during agricultural activities and sustained periods of normal farmyard activities for which dust levels may be somewhat higher than ambient. Various sources have indicated a range of dust-loading factors for agricultural and construction activities from 0.01 to 1.8 mg/m^3 [IAEA, 1987; Oztunali, et al., 1981; and Yu, et al., 1993]. For comparison, the Occupational Health and Safety Administration (OHSA) limit in 29 CFR 1910.1000 for the respirable fraction of dust without use of a respirator is 5 mg/m^3. The respirable fraction of the baghouse dust is unknown. Therefore, the staff conservatively assumed the upper bound on the dust loading factor to be 1.8 mg/m^3. The staff also assumed 100% to be respirable.

The exposure time to airborne baghouse dust was based on conversations with the licensee. SMC indicated in a telephone conversation with NRC staff, on July 7, 1997, that the inventory of baghouse dust could be processed at the cement manufacturer within 4 months (16 weeks). Therefore, the staff assumed that the worker would be exposed 40 hours per week for 16 weeks.

The external dose was calculated based on the licensee's calculation of exposure in a letter dated June 24, 1996. The licensee calculated an exposure of 93 mR/hr at 1 meter above an infinite plane of baghouse dust 1-meter-thick using the Microshield 4.21 code. The staff confirmed that this exposure rate was not an unreasonable estimate. SMC assumed an exposure time of one hour per week. However, to be conservative, the staff assumed that the worker would be exposed up to 10 hours per week.

Based on these assumptions, the estimated total effective dose equivalent to the maximally exposed individual is not expected to exceed of 1.30 mSv/yr (130 mrem/yr) based on the old methodology of ICRP Publications 26 [1977] and 48 [1986]. Using the new methodology based on ICRP Publications 66 [1994] and 68 [1994] the dose to the maximally exposed individual is not expected to exceed 0.35 mSv/yr (35 mrem/yr), and if more realistic assumptions are considered, the maximum dose is not expected to exceed 0.02 mSv/yr (2 mrem/yr). Therefore, based on this analysis it is not expected that a member of the public will receive a dose greater than 1 mSv/yr (100 mrem/yr) from these operations.

References

1. U.S. Nuclear Regulatory Commission, Memorandum from Sherry C. Wu, NRC, to Michael F. Weber, NRC, August 7, 1997.

2. International Commission on Radiological Protection, "Recommendations of the International Commission on Radiological Protection," ICRP Publication 26, Annals of the ICRP, Vol. 1, No. 3 (Pergamon Press, New York), 1977.

3. International Commission on Radiological Protection, "The Metabolism of Plutonium and Related Elements," ICRP Publication 48, Annals of the ICRP, Vol. 16, No. 2-3 (Pergamon Press, New York), 1986.

4. International Commission on Radiological Protection, "Human Respiratory Tract Model for Radiological Protection," ICRP Publication 66, Annals of the ICRP Vol. 24, No. 1-3 (Pergamon Press, New York), 1994.

5. International Commission on Radiological Protection, "Dose Coefficients for Intakes of Radionuclides by Workers," ICRP Publication 68, Annals of the ICRP Vol. 24, No. 4 (Pergamon Press, New York), 1994.

6. Letter from D.R. Smith, SMC, to J.D. Kinneman, USNRC, May 13, 1998.

7. National Council on Radiation Protection and Measurements, "Exposures from the Uranium Series with Emphasis on Radon and its Daughters," NCRP Report No. 77, 1984.

8. Yu, C. et al., "Data Collection Handbook to Support Modeling the Impacts of Radioactive Material in Soil," ANL/EAIS-8 (Argonne National Laboratory), April 1993.

9. Gilbert, T.L., et al., 1983, "Pathways Analysis and Radiation Dose Estimates for Radioactive Residues at Formerly Utilized MED/AEC Sites," ORO-832 (Rev.), prepared by Argonne National Laboratory, Argonne, IL., for U.S. Department of Energy, Oak Ridge Operations, Oak Ridge, Tenn., March (reprinted with corrections January 1984).

10. International Atomic Energy Agency (IAEA), "Exemption of Radiation Sources and Practices from Regulatory Control," Interim Report, IAEA-TECDOC-401, Vienna, Austria, 1987.

11. Oztunali, O.I., et al., "Data Base for Radioactive Waste Management, Impacts Analyses Methodology Report, NUREG/CR-1759, Vol. 3, prepared by Dames and Moore, White Plains, NY, for the Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C., 1981.

12. Phone conversation between C.S. Eves of Shieldalloy Metallurgical Corporation and H.M. Astwood and S.C. Wu of the USNRC, July 7, 1997.

13. Grove Engineering, Inc., Microshield 4.21, February 1995.

ATTACHMENT 2

DRAFT

Mr. David R. Smith
Radiation Safety Officer
Shieldalloy Metallurgical Corporation
P.O. Box 768
Newfield, New Jersey 08344

SUBJECT: TRANSFER OF BAGHOUSE DUST UNDER 10 CFR 40.13(A) AND 40.51(b)(3) (TAC NO. L30926)

Dear Mr. Smith:

The U.S. Nuclear Regulatory Commissioin (NRC) staff has reviewed your request to transfer baghouse dust containing less than 0.05 weight percent source material to an exempt person, in accordance with 10 CFR 40.51(b)(3) and 40.13(a). The staff also reviewed your report entitled, "Technical Basis for the Use of Baghouse Dust as an Additive in Concrete Production." Based on the information you submitted, the NRC performed independent dose calculations to determine the potential doses to individual members of the public as a result of these transfers. The staff has determined, based on the results of these assessments that the total effective dose equivalent to an individual member of the public or worker from the transfer of the material to a cement manufacturer is not expected to exceed 100 mrem/yr specified in 10 CFR 20.1301(a), under routine conditions. Based on this determination, the NRC staff has no objection to you transferring this material.

If you have any questions about this letter or the assessments that were performed, please call Heather Astwood of my staff, on (301) 415-5819.

Sincerely,
Carl Paperiello, Director
Office of Nuclear Material Safety and Safeguards

Docket 40-7102
License SMB-743

Enclosure: As stated

DRAFT

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1. It should be noted that there is NRC precedent (Health Physics Position-190, NUREG/CR-5569, Rev. 1) to deny requests to transfer material under the 10 CFR 40.13(a) limit for the purposes of unregulated waste disposal. Waste disposal by transfer to a person licensed for waste disposal is permitted by 10 CFR 20.2001. However, an Office of the General Counsel (OGC) position is that an exempt person is not authorized to receive radioactive waste for disposal except through the application of 10 CFR 20.2002 (e.g., the cesium-137 contaminated baghouse dust policy). Generally, persons performing disposal activities for radioactive waste coming from licensed generators need to be licensed.

2. Note that 10 CFR Part 20 has not been updated to reflect the new ICRP methodology. However, NUREG-1549, "Demonstrating Compliance with the Radiological Criteria for Decommissioning," indicates that licensees may use this methodology when implementing the recently promulgated license termination rule in 10 CFR Part 20, Subpart E. NUREG-1549 has been approved by the Commission for interim use [SECY Suspense 7/31/98].

3. The SMC submittal dated June 5, 1997, states in a footnote that the "Th/U activity ratio is 4.5 - 1.4." However, calculations in the submittal use this value as the ratio of Th-232 to U-238, not as the ratio of total thorium (Th-232 + Th-228 + Th-234 + Th-230 + Th-231 +Th-227) to total uranium (U-234 + U-235 + U-238). A Th-232/U-238 ratio of 4.5 results in Th(total)/U(total) ratio of 5.37.